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Nuclear Waste Management Technical Support in the Development of Nuclear Waste Form Criteria for the NRC

Task 3
Waste Inventory Review

ORIGINAL

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Prepared for
U.S. Nuclear Regulatory Commission

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All references made to the BNL interim reports, "NRC Nuclear Waste Management Technical Support in the Development of Nuclear Waste Form Criteria," Tasks 1-5, should be disregarded. The following references should be inserted in their places, as applicable.

R. Dayal, et al., "Nuclear Waste Management Technical Support in the Development of Nuclear Waste Form Criteria for the NRC, Task 1: Waste Package Overview," NUREG/CR-2333, Vol. 1, BNL-NUREG-51458, February 1982.

G. Bida, et al., "Nuclear Waste Management Technical Support in the Development of Nuclear Waste Form Criteria for the NRC, Task 2: Alternative TRU Technologies," NUREG/CR-2333, Vol. 2, BNL-NUREG-51458, February 1982.

D. Majumdar, et al., "Nuclear Waste Management Technical Support in the Development of Nuclear Waste Form Criteria for the NRC, Task 3: Waste Inventory Review," NUREG/CR-2333, Vol. 3, BNL-NUREG-51458, February 1982.

T.M. Ahn, et al., "Nuclear Waste Management Technical Support in the Development of Nuclear Waste Form Criteria for the NRC, Task 4: Test Development Review," NUREG/CR-2333, Vol. 4, BNL-NUREG-51458, February 1982.

M.S. Davis, "Nuclear Waste Management Technical Support in the Development of Nuclear Waste Form Criteria for the NRC, Task 5: National Waste Package Program," NUREG/CR-2333, Vol. 5, BNL-NUREG-51458, February 1982.

PART 1

INVENTORY OF HIGH LEVEL AND TRANSURANIC WASTES IN THE UNITED STATES

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ABSTRACT

This inventory of high level nuclear waste in the U.S. was prepared by the Nuclear Waste Management Division of Brookhaven National Laboratory for the Office of Nuclear Materials Safety and Safeguards of the U.S. Nuclear Regulatory Commission. Existing and projected inventories of high level waste (HLW), transuranic (TRU) waste, and spent fuel will be required in making prudent decisions in the licensing of high level waste repositories. This report describes the quantity and characteristics of existing and projected HLW and TRU waste from the government nuclear programs, the existing HLW and TRU waste from the commercial power industry, and the existing and projected spent fuel from the commercial nuclear power industry.

CONTENTS

ABSTRACT iii
FIGURES vi
TABLES vii
ACKNOWLEDGEMENTS xii
EXECUTIVE SUMMARY. xiii

1. INTRODUCTION 1

2. HIGH LEVEL RADIOACTIVE WASTE INVENTORY 3
   2.1 High Level Waste at West Valley 3
      2.1.1 Neutralized Waste in Tank 8D2 6
      2.1.2 Acidic Waste in Tank 8D4 14
   2.2 DOE High Level Radioactive Waste 14
      2.2.1 High Level Waste at Idaho National Engineering Laboratory 14
      2.2.2 High Level Waste at Savannah River Plant 40
      2.2.3 High Level Waste at Hanford Reservation Operations . . 59
   2.3 Summary 81
   2.4 References 84

3. EXISTING COMMERCIAL SPENT FUEL INVENTORY 87
   3.1 References 92

4. PROJECTED COMMERCIAL SPENT FUEL INVENTORY 93
   4.1 Introduction 93
   4.2 Discussion of Spent Fuel Projections 94
   4.3 Projected Spent Fuel Inventory 105
   4.4 References 109

5. EXISTING AND PROJECTED TRU INVENTORY 111
   5.1 Introduction 111
   5.2 Description of TRU Wastes 111
   5.3 Inventories of DOE TRU Waste 115
      5.3.1 Buried TRU Waste 115
      5.3.2 Stored TRU Waste 115
      5.3.3 Additional TRU Waste Data for the Idaho National Engineering Laboratory (INEL) Site 118
   5.4 Projections of TRU Wastes at DOE Sites 118
   5.5 Inventories of Commercial TRU Waste 118
   5.6 References 125

APPENDIX A - SCOPE OF WORK 127
APPENDIX B - GLOSSARY OF SELECTED TERMS 128
APPENDIX C - CHARACTERISTICS OF IMPORTANT RADIONUCLIDES 132
APPENDIX D - LIST OF COMPUTER PROGRAMS EMPLOYED IN RADIOACTIVE WASTE INVENTORY ASSESSMENTS 141
APPENDIX E - CHARACTERISTICS OF COMMERCIAL SPENT FUEL 143

FIGURES

2.1 Volume and distribution of waste in Waste Tank 8D2 at West Valley. 8

2.2 Idaho high level waste inventory from 1964-1974 26

2.3 High level defense waste total curie content 34

2.4 High level defense waste thermal histories 35

2.5 Significant actinides in ICPP calcine 39

2.6 Schematic diagram of waste processing in-tank farm at SRP: (a) high-heat waste (b) low-heat waste. 41

2.7 Savannah River Plant high level waste volumes - F Area 44

2.8 Savannah River Plant high level waste volumes - H Area 45

2.9 Savannah River high level waste inventory from 1961-1974. . . . 46

2.10 Radionuclide composition of SRP fresh waste 52

2.11 Radionuclide composition of SRP reconstituted waste: 0 to 1400 years after irradiation 55

2.12 Radionuclide composition of SRP reconstituted waste: 1 to 10^6 years after irradiation 56

2.13 Hanford radioactive waste management. 60

2.14 Volumes of high level waste at Hanford from 1945-1985 64

2.15 Radionuclide content of Hanford in-tank waste 72

2.16 Radionuclide content of Hanford encapsulated waste. 73

4.1 Nuclear generating capacity projections 95

TABLES

2.1 Distribution of Radioactivity at Western New York Nuclear Service Center Facilities. 4

2.2 Summary of Fuel Receipts at the NFS Reprocessing Plant (1972). . 5

2.3 Source and Quantity of Fuel Pool Contents at West Valley (as of May 31, 1976). 7

2.4 Contents of Waste Tank 8D2 in West Valley From NFS Waste Surveillance Program Quarterly Report to New York State ERDA . . 10

2.5 Calculated Radioactivity Inventory and Decay Heating Rates in the Neutralized Tank at West Valley. 11

2.6 Calculated Actinide Element Inventory in Tank 8D2 at West Valley 12

2.7 Estimated Composition of Contents of Waste Tank 8D2 at West Valley 13

2.8 Estimated Constitution of Thorex Waste in Tank 8D4 at West Valley 15

2.9 High-level Liquid Waste in the Acidic Tank at West Valley From Waste Surveillance Program 15

2.10 ORNL Analysis of Acidic Waste at West Valley (September 23, 1978) 16

2.11 Radioactive Liquid Waste Inventory in Different Tanks at ICPP. . 18

2.12 Chemical Composition of High Level Zirconium Fluoride Wastes at ICPP. 20

2.13 Chemical Composition of High Level Non-Fluoride Wastes at ICPP 20

2.14 Chemical Composition High Level Aluminum Nitrate Waste at ICPP 21

2.15 Chemical Composition of Intermediate Level Wastes at ICPP . . . 22

2.16 Average Composition of High Level Liquid Waste at ICPP 22

2.17 Radiochemical Analysis of High Level Zirconium Fluoride Wastes at ICPP 23

2.18 Radiochemical Analysis of High Level Non-Fluoride Wastes at ICPP 23

2.19 Radiochemical Analysis of High Level Aluminum Nitrate Waste at ICPP 24

2.20 Radiochemical Analysis of Intermediate Level Wastes at ICPP Tanks 24

2.21 Actinide Analysis of Stored ICPP First Cycle Raffinate 25

2.22 Decay Heat Generation Rates in ICPP Tanks 25

2.23 Inventory of Calcine in Storage at ICPP 27

2.24 Compositions Representative of ICPP Raw Waste 29

2.25 Typical Properties of Calcined Solids at ICPP. 30

2.26 Composition of ICPP Calcines 31

2.27 Radionuclide Content of INEL Defense High Level Waste in Calcine Form 31

2.28 INEL Defense High Level Calcine Waste Curie Content and Heating Rates as a Function of Time 33

2.29 Chemical Composition of Retrieved Alumina and Zirconia Calcines at ICPP 36

2.30 Radiochemical Analysis of Retrieved Alumina Calcine From Bin 3 at Idaho 37

2.31 Radiochemical Analysis of Retrieved Zirconia Calcine From Bin 7 at Idaho 38

2.32 Activity of Calcined Solids by Major Radionuclides 40

2.33 Existing and Projected Volumes of High Level Waste at Savannah River Site 43

2.34 General Composition of SRP Sludge 43

2.35 Average Chemical Composition of SRP High-Level Waste 47

2.36 Ranges of Chemical Composition of SRP Supernates 48

2.37 Concentration Range of Major Constituents in SRP Supernate . . . 48

2.38 Chemical Composition of SRP Sludges. 48

2.39 SRP Tank 15 Sludge Analysis 49

2.40 Major Components in SRP Waste Calcines 49

2.41 SRP Tanks 19 and 20 Salt Cake Analysis 50

2.42 Average Radionuclide Composition of Fresh SRP High-Level Waste . 51

2.43 Average Radionuclide Composition of SRP High-Level Supernate . . 53

2.44 Average Radionuclide Composition of SRP High-Level Sludge. . . . 54

2.45 Radionuclide Content of Reconstituted SRP High-Level Waste (1985) 57

2.46 Radionuclide Content of Savannah River Defense High Level Waste in Glass Form 57

2.47 Savannah River Defense High Level Waste Curie Heat Contents . . 58

2.48 Hanford Reactor Status Summary 59

2.49 Radioactive Waste Sources at Hanford 62

2.50 Hanford High-Level Waste Storage Tanks 63

2.51 Estimates of HLW Quantities at Hanford 65

2.52 Total Mass (kg) of In-Tank Existing High Level Fission Products at Hanford as a Function of Time After 1990 65

2.53 Mass of Existing In-Tank Waste Elements in Hanford as of 1995. . 66

2.54 Inventory of Major Fission Products in Hanford High Level Waste Decayed to 1990 67

2.55 Total Activity (in Millions Curies) of In-Tank Existing High Level Fission Products at Hanford as a Function of Time After 1990 68

2.56 Existing Total Mass (kg), Activity (Ci) and Thermal Power (w) of In-Tank Actinides at Hanford as a Function of Time After 1990 69

2.57 Estimated Inventory and Characteristics of Solids Stored In Tanks, Year 1980 70

2.58 Total Thermal Power (kw) of In-Tank Existing High Level Fission Product Wastes at Hanford as a Function of Time After 1990 70

2.59 Major Chemicals in Hanford High Level Defense Waste In-Tank Inventory 74

2.60 Average Composition of Solids Stored in Tanks at Hanford (1980). 75

2.61 Concentrations of Hanford High-Heat Wastes 75

2.62 Concentrations of Hanford Low-Heat Wastes. 76

2.63 Radionuclide Content of Hanford Defense High Level Waste in Glass Form 76

2.64 Hanford Defense High Level Waste Activity and Heating Rate . . . 77

2.65 Estimated Hanford High Level Waste in 1990 79

2.66 Projected Volumes of High Level Defense Wastes to be Accumulated 80

2.67 Summary of U. S. High Level Nuclear Wastes 82

3.1 Existing Commercial Spent Fuel Inventory as of December 31, 1979 88

3.2 NFS West Valley Source and Quantity of Fuel Pool Contents (Updated to February 26, 1980) 91

3.3 Commercial Fuel Reprocessed at NFS 91

4.1 Forecast of U.S. Nuclear Electric Power Generating Capacity, GWE, Used in ORNL/TM-6008. 101

4.2 Reference and Low-Growth Projections From DOE/ET-028. 102

4.3 SMSC % of Completion Model for DOE/ET-0075 104

4.4 Nuclear Generating Capacity Installed and Discharging Spent Fuel for Each Year, 1979 - 2000 From NUREG-0575 106

4.5 Annual and Cummulated Schedules of Spent Fuel Discharge From NUREG-0575 107

4.6 DOE/EIA Mid Case Projection, July 1980 108

4.7 Yearly Discharges and Spent Fuel Accumulations, Mtu, (From ORO-778) 108

5.1 Volumes and Pu Contents for Important Waste Forms Stored at Idaho from September 1971 Through December 1979 112

5.2 RFP and LASL Sludge Analysis. 112

5.3 Typical Incinerator Feed Material 113

5.4 Residue Compositions From Selected Incinerators 114

5.5 Transuranic Content of Idaho and Savannah River TRU Waste in Retrievable Storage 114

5.6 Inventories of TRU Waste Buried at DOE Sites 116

5.7 Inventories of TRU Waste Retrievably Stored at DOE Sites 117

5.8 Radionuclide Content of Buried and Stored TRU Waste at INEL . . 119

5.9 Estimated Activity in Beta-Gamma Waste Mixed With Buried TRU Waste at INEL 120

5.10 Projected Storage of Retrievable Transuranic Defense Waste at INEL 121

5.11 Projected Storage of Retrievable Transuranic Defense Waste at Hanford 121

5.12 Projected Storage of Retrievable Transuranic Defense Waste at SRP 122

5.13 Projected Storage of Retrievable Transuranic Defense Waste at LASL 122

5.14 Projected Storage of Retrievable Transuranic Waste at ORNL . . . 123

5.15 Projected Storage of Retrievable Transuranic Waste at NTS . . . 123

5.16 Cumulative Waste Inventory at Commercial Burial Sites (1978) . . 124

5.17 Projected Generation of Industrial Transuranic Wastes 125

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EXECUTIVE SUMMARY

This inventory of high level nuclear waste in the U.S. was developed from existing data, reports, and other sources of information. Existing and projected inventories of high level waste (HLW), transuranic (TRU) waste, spent fuel, and decommissioning and dismantling (D&D) waste will be required in making prudent decisions in the licensing of high level waste repositories.

Numerous references, reports, and data sources were consulted for this work. Discussions with responsible individuals at the major waste storage facilities were held to help resolve and understand discrepancies that appear in the literature. Low level wastes are excluded from this inventory. For decommissioning and dismantling wastes, no detailed projection of future wastes could be developed although rough D&D waste projections for commercial power reactors are available.

The inventory is reported in three parts. Part 1 gives the existing and projected inventory of HLW, spent fuel, and TRU wastes. Part 2 is concerned with the characteristics and volume of high activity D&D waste from commercial power reactors. Part 3 provides additional information on D&D wastes from power reactors and particle accelerators.

The major findings of the reports may be summarized as follows:

- A reasonable estimate of the total inventory of existing HLW and commercial spent fuel is available in terms of the volume, average range of chemical composition, and radioactivity.

- The accuracy of certain inventory estimates and the methods used to estimate the quantitites and characteristics of the waste are not all known at this time. There are also certain discrepancies that cannot be resolved without additional study and measurement.

- There are about 75 million gallons and about 1300 million curies of liquid sludge, salt cake, and other forms of HLW at the government facilities. About 40% of the total volume of HLW is in a liquid form. The Idaho, Savannah River, and Hanford facilities contain, respectively, about 3%, 30%, and 67% of the total waste volume and about 7%, 43%, and 45% of the waste radioactivity in curies. The total waste and the percentages among the three varies from year to year, and the data given above are for 1978.

- A considerable number of irradiated fuel elements awaiting processing are stored at government facilities. The exact number and characteristics of all these elements were not determined for this inventory.

- There are about 5200 metric tons of heavy metal (MTHM) in the form of spent fuel assemblies stored at commercial power reactors and

away-from-reactor storage facilities at the end of 1979. Based upon 5.3 x 10^5 Ci/MT for 5-year-old fuel, the estimated activity would be in the range of 2500-3000 million curies.

- The most reasonable projection of nuclear electric generating capacity reaches about 180 GWE by the year 2000. This will result in an accumulation of about 89,000 MTHM in the form of spent fuel assemblies in the year 2000.

- There were about 319,000 m^3 of TRU waste buried at DOE sites as of October 1979. The quantity of TRU elements contained in this waste is estimated to be about 785 kg. About 55,000 m^3 of TRU waste is retrievably stored at DOE facilities. This waste contains about 685 kg of TRU elements.

- High activity D&D wastes from commercial nuclear power reactors are primarily neutron activated reactor components such as the reactor vessel and internals. For a large 1175 MWe PWR, the total mass of such components is estimated to be about 600 tonnes and the radioactivity level about 5 x 10^6 curies. For a BWR of 1155 MWe, the mass of activated components estimated to be about 300 tonnes and the radioactivity level about 6.5 x 10^6 curies. These wastes are important in regard to waste management since they may contain long lived activation products and cannot be easily decontaminated.

- Only rough estimates are available for D&D wastes from other nuclear facilities, such as particle accelerators. Decommissioning and dismantling waste volumes from surplus DOE facilities are reported to be approximately 9.5 x 10^3 m^3 of TRU wastes and 1.1 x 10^5 m^3 of non-TRU wastes by the year 2000. However, most of these are believed to be low activity wastes.

The characteristics of HLW and TRU wastes, such as the chemical composition, physical form, and radionuclide content, are important in analyzing the behavior of candidate waste forms for long term isolation. When appropriate, the characterization of HLW and TRU wastes should be determined with greater certainty than is now available.

INVENTORY OF HIGH LEVEL NUCLEAR WASTES IN THE UNITED STATES

1. INTRODUCTION

This report gives an account of the inventory of high level waste (HLW) and transuranic (TRU) waste within the United States. The data contained in this report is an inventory of the following wastes:

Government HLW Waste (Existing and Projected)
Existing Commercial HLW
Existing Commercial Spent Fuel
Projections of Commercial Spent Fuel
Existing and Projected TRU Waste

The Scope of Work for this inventory task is contained in Appendix A. Appendix B is a glossary of selected terms used in describing the nuclear waste inventory. Appendix C contains data on the decay characteristics and production modes of radionuclides of interest to nuclear waste management. Appendix D briefly describes some computer codes and systems used in inventory assessments. Appendix E gives the characteristics of commercial spent fuel as reported in the Oak Ridge Report ORNL/TM-6008. For purposes of this inventory the following definitions are used:

High Level Waste

For the once through fuel cycle the spent fuel assemblies constitute the HLW. HLW other than spent fuel arises from reprocessing of government owned fuels and a small amount of commercial fuel reprocessed in the past. For these cases the waste consists of the following wastes generated at the fuel reprocessing plant:

- the nitric acid solution from the first solvent extraction cycle
- the solids and sludge from the feed clarification step
- the aqueous waste concentrator residues from the U-Pu partition step
- second and third cycle raffinates from government reprocessing plants.

Transuranic Wastes

Wastes contaminated with transuranic nuclides such that the radioactivity is equal to or greater than 10 nanocuries per gram (10 nCi/g) or wastes suspected of being contaminated to these levels.

For the once through fuel cycle the following are the sources of TRU waste:

- the spent fuel assemblies
- suspect TRU filter sludges from BWR water polishing systems[a]

[a]It is not clear at this time if these components are to be considered or treated as TRU waste.

- suspect TRU wastes from fuel assembly hardware, in-core instruments, control rods, etc.[a]

Most TRU wastes have been generated in defense and research programs at the major DOE facilities.

The inventory data contained in this report, in accordance with the Scope of Work, is derived from unclassified published sources, facility records, and discussions with certain individuals familiar with specific waste inventory data. The inventory data contained in the following sections of this report will be briefly described.

Section 2 gives inventory data on existing and projected quantities of government high level waste and existing commercial high level waste. The term "government waste" refers to HLW generated by government agencies, primarily the Atomic Energy Commission and its successors ERDA and DOE, in various nuclear programs. The major contributor to the existing inventory of HLW is the nuclear weapons program that began as the Manhattan Project. Most of these wastes have been generated from the reprocessing of spent fuels to extract weapons grade plutonium. This type of activity has occurred primarily at the Hanford Project and the Savannah River Plant. Another source of HLW is the reprocessing of naval reactor and research reactor fuels. These fuels are generally composed of alloys or special compositions in which the uranium content is substantially enriched in the U-235 isotope. These fuels are reprocessed at the Idaho National Engineering Laboratory. The only other major source of HLW in the U. S. is at the Nuclear Fuel Services, West Valley, New York reprocessing plant which reprocessed commercial (and some government owned) spent fuel from about 1966 to 1972. Under present U. S. policy the reprocessing of commercial spent fuel has been deferred indefinitely although reprocessing in the government weapons, R&D, and naval programs is expected to continue and where available projections of these quantities of HLW have been noted. It has been difficult to ascertain the accuracy of the data concerning these HLW inventories. Knowledge of such data as the chemical composition, radionuclide content, heat generation rate and other parameters is of vital importance in assessing the suitability of proposed waste processing and packaging methods. This is particularly true where the waste form is sensitive to the character of the waste load. Hence it may be advisable to have an accurate characterization of the waste early in order not to delay a decision on licensing. However, this must be balanced against the possibility that characterization now may not be helpful in predicting the character of the waste at the time of actual processing and packaging.

Section 3 describes the inventory of commercial spent fuel in storage as of December 31, 1979. This has been obtained from published sources based on data supplied by licensed utilities. Data on the spent fuel from older government owned or other special reactors, where available, have also been included.

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[a]It is not clear at this time if these components are to be considered or treated as TRU waste.

Section 4 contains a fairly comprehensive discussion on projections of commercial spent fuel to the year 2000. The inventory projection selected for this report places the cumulative commercial spent fuel inventory in the year 2000 at about 89,000 metric tons of heavy metal (MTHM).

Section 5 contains data on TRU wastes from both the government and private sectors. Existing quantities of both buried and retrievably stored wastes are given as well as projections to the year 2000. The uncertainty concerning the volumes of buried waste is considerably higher than that for retrievably stored waste. The volume of buried waste is believed to be in excess of 320,000 m^3 while the volume of retrievably stored waste is about 55,000 m^3.

A draft version of this report was published in April 1980. While the present report contains the most recent data available, the draft may be consulted as a handy compilation of past inventory data sources.

2. HIGH LEVEL RADIOACTIVE WASTE INVENTORY

The high level radioactive waste inventory reported in this section describes the wastes generated from the reprocessing of spent fuels from government nuclear programs and a portion of the commercial power generating industry. The commercial HLW was generated at the Nuclear Fuel Services, West Valley New York, facility from about 1966 to 1972, and the inventory of these wastes are reported below in Section 2.1. The government or DOE high level wastes resulted primarily from the reprocessing of fuels from Pu production reactors, naval propulsion reactors, and a variety of research and test reactors. These programs began in the early and mid 1940s and are expected to continue for some years to come. The Idaho National Engineering Laboratory has been primarily involved in reprocessing naval and test reactor fuels. The Idaho inventory is presented in Section 2.2.1. The HLW at the Hanford and Savannah River sites is primarily from the reprocessing of Pu production reactor fuels as part of the nuclear weapons program. The Savannah River Plant inventory is presented in Section 2.2.2 and the Hanford inventory is presented in Section 2.2.3.

2.1 High Level Radioactive Waste at West Valley

The radioactive waste in West Valley may be divided into four separate categories: High level liquid waste, high level solid waste in NRC licensed burial areas, low level waste in state licensed burial areas and radioactivity left in the reprocessing plant. Types of wastes accumulated in each of these categories, their volumes and their overall activities are shown in Table 2.1, which shows that liquid high level waste is the most important waste at West Valley.[1] The NRC licensed waste burial area began operation in 1966, and was used, until 1972, for disposal of high level radioactive solid wastes generated by the reprocessing plant. It also has 42 ruptured spent fuel elements from the Hanford N-reactor. Since 1972, this area is used as a burial ground for low level radioactive wastes.

A list of fuel reprocessed at West Valley is given in Table 2.2. It includes the names of the reactors which produced the fuel and the amount of

Table 2.1

Distribution of Radioactivity at Western New York Nuclear Service Center Facilities

Waste Category	Type of Waste Stored	Size of Tank/Area	Volume Stored	Curies of Activity[1] Short Lived	Long Lived[a]
Neutralized liquid waste (Underground Tank 8D2)	Waste generated from normal operation of the NFS reprocessing plant using the Purex process.	750,000 gal carbon steel tank	561,000 gal (2,120,000 L)	39×10^6	2.3×10^4
Acidic liquid waste (Underground Tank 8D4)	Waste generated from processing Th-U fuel from Indian Point-1 reactor using the Thorex process.	15,000 gal stainless steel tank	10,964 gal[b] (41,500 L)	2.4×10^6	3.6
NRC-licensed burial area[c]	Highly radioactive solid and other wastes generated by the reprocessing plant including the hull or fuel hardware.	7.2 acres	3900 m^3	0.55×10^6	550
State-licensed burial area[c]	Low level radioactive waste generated at West Valley and from off-site sources (through 1972).	22 acres	48,565 m^3[1]	0.71×10^6	142
Reprocessing plant[d]	Plant inventory of radioactivity - processing cells, support areas, etc.	-----	-----	0.012×10^6	120
TOTAL				43×10^6	2.4×10^4

[a]Long lived = greater than 100-year half life.
[b]Measured by probe (1979).
[c]Activity at time of burial.
[d]Data provided by Nuclear Fuel Services, Inc.

Table 2.2

Summary of Fuel Receipts at the NFS Reprocessing Plant (1972)[2,3]

Reactor	Fuel Type	Location	MTU	KGPU
Big Rock Point Consumers Power Co.	BWR	Charlevoix, Michigan	24.2	100.7
Yankee-Rowe Yankee Atomic Electric Co.	PWR	Rowe, Massachusetts	88.2	652.4
Indian Point-1 Consolidated Edison Co.	Th-U core	Buchanan, New York	39.2[a]	175.7
Humbolt Bay Pacific Gas & Electric Co.	BWR	Eureka, California	20.8	87.2
Bonus Puerto Rico Water Authority	BWR	Puerto Rico	4.1	4.9
Pathfinder Northern States Power	BWR	Sioux Falls, South Dakota	9.6	7.1
Dresden-1 Commonwealth Edison Co.	BWR	Morris, Illinois	71.3	295.6
CVNPA	HWR	Parr, South Carolina	3.5	11.6
NPR	-----	Richland, Washington	379.4	553.0
Sefor Southeast Utilities	Fast reactor U-Pu oxide	Fayetteville, Arkansas	0.4	95.5
			640.7	1983.7

[a]Includes 16 MT thorium + U (85% ^{235}U at discharge).

uranium and plutonium it contained.[2,3] The amount and content of fuel in the fuel storage pool are given in Table 2.3. Spent fuel was last received at the plant in 1975. The heat from the 157 MTU of spent fuel currently in the pool is cooled by coolers with a cooling capacity of 3.5 MW.

There are two types of high level liquid waste stored at West Valley. Most of the stored waste was generated from the normal operation of the NFS reprocessing plant to reprocess uranium fuel using the Purex process. This waste, neutralized by the addition of sodium hydroxide, is contained in a large carbon steel tank designated as Tank 8D2. It has a supernatant liquid and a sludge at the bottom. The second type of waste came from processing a small batch of thorium-uranium fuel from the Indian Point-1 reactor using the Thorex process. This acidic waste (thorium would otherwise precipitate) is stored in a cooled stainless steel tank designated as Waste Tank 8D4. The waste is assumed to remain as a solution in this tank and if there is a sludge it is expected to be small. There are two other spare tanks 8D1 and 8D3 available at West Valley for emergency purposes.

Before proceeding with the description of the high level liquid waste in the two tanks, it is important to realize the difficulties encountered in documenting an accurate inventory of materials in large storage tanks. Most of these are probably problems generic to all neutralized radioactive waste storage sites and are listed below:

- Large volume.
- Composition of sludge may vary both radially and vertically; almost impossible to define the sludge-supernate interface.
- Volume of sludge unknown.
- Relative quantity of supernate and sludge unknown.
- Composition of sludge unknown.
- Sludge may have several layers with different densities in each layer; the bottom dense layer is likely to slope downward away from the inlet point of the tank; Figure 2.1 gives a description of possible waste composition in Tank 8D2.
- Almost impossible to obtain a good representative sample of sludge; hence interpretation of any sample is subject to large uncertainties.
- Accuracy of sample measurements for constituents is in the range of ±20%.
- Except for determining cesium activity in supernate there are no routine analyses of fission products in Tank 8D2.
- Unspecified small percentage of fuel cladding is dissolved in the solution in Tank 8D4.

The high level waste inventory at West Valley is discussed separately for the two tanks:

2.1.1 Neutralized Waste in Tank 8D2

Total volume of waste[4] = 561,000 gallons (= 2.12 million liters). An earlier NRC report, NUREG-0043,[5] quotes the total volume to be 600,000

Table 2.3

Source and Quantity[1] of Fuel Pool Contents[a] at West Valley (as of May 31, 1976)

Utility	Reactor	Fuel Size	Number of Assemblies	Number of Canisters	Amount, MTU
Commonwealth Edison	Dresden (BWR)	10.2 cm sq. 340 cm long	206	69	20.435
Rochester Gas & Electric	Ginna (PWR)	19.72 cm sq. 406.9 cm long	121	121	46.156
Consumers Power	Big Rock Point (BWR)	16.5 cm sq. 208-213 cm long	85	85	11.130
Wisconsin Electric & Power Co. (WEPCO)	Point Beach 1 & 2 (PWR)	19.72 cm sq. 405.1 cm long	120	120	45.298
General Public Utilities (GPU)	Oyster Creek (BWR)	13.9 cm sq. 439 cm long	183	183	33,987
Sub-Totals			715	578	157.006
Rochester Gas & Electric	Ginna	Neutron clusters Control rods		1 2	
Commonwealth Edison	Dresden	Thoria rods		1	
TOTAL			715	582	157.006

[a]Information provided by Nuclear Fuel Services, Inc.

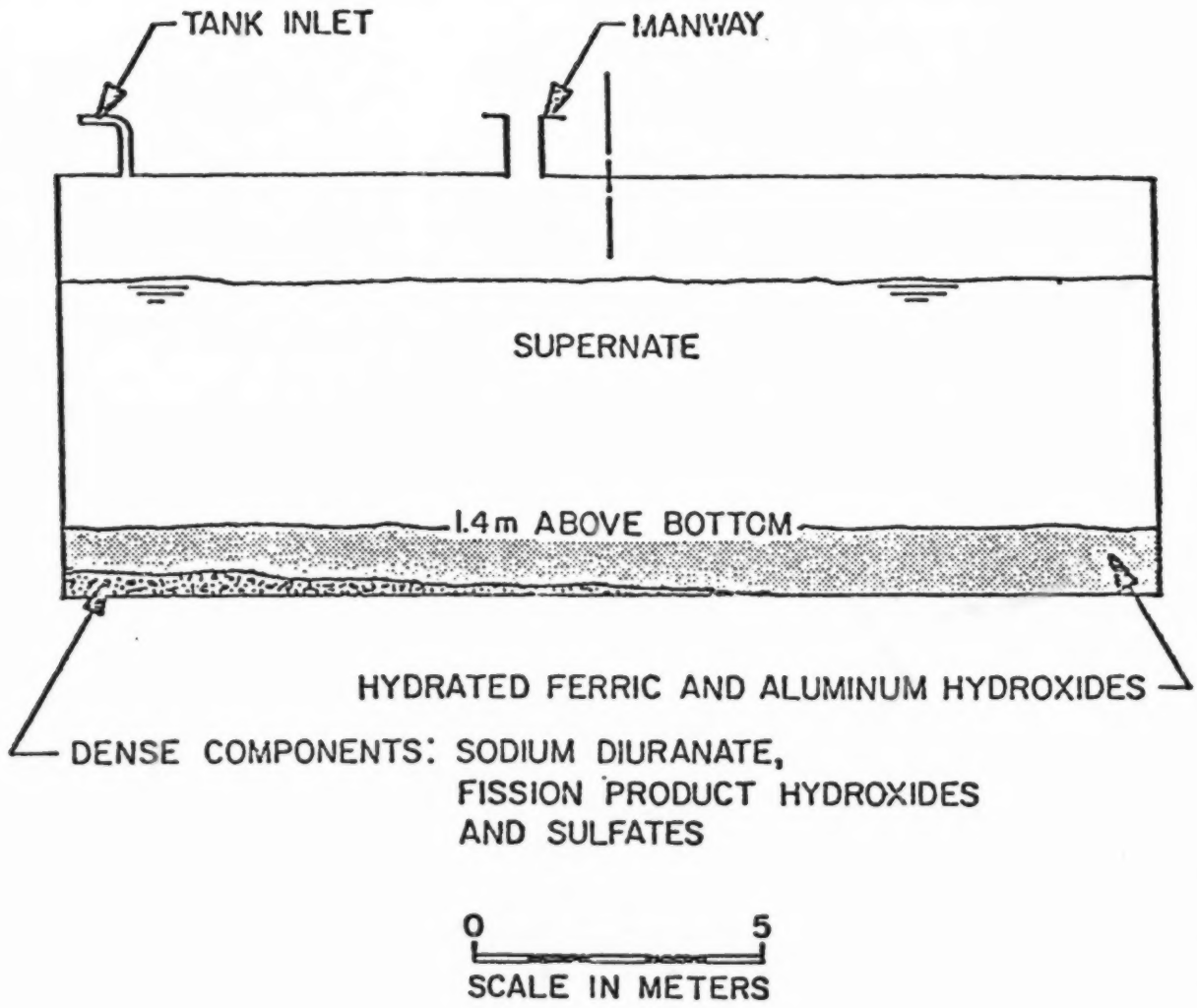

Figure 2.1. Volume and distribution of waste in Waste Tank 8D2 at West Valley.[1]

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gallons. The IRG report[6] also states this volume to be 600,000 gallons or 80.2 thousand cubic feet as of 10/1/77. These documents have probably rounded off the numbers.

No good data for the volume of the sludge are available. In fact, the numbers quoted vary widely. Reference 5 states that the sludge volume is 30,000 gallons (= 114,000 liters). Rough bottle-drop measurements by NFS[2] suggest that the sludge at the bottom of the tank is 1.4 to 1.5 m deep. This would then indicate[1] a volume of about 120,000 gallons (= 454,000 liters). This is, however, complicated by the fact that the sludge height is not uniform as shown in Figure 2.1. Reference 1 suggests that there is a minimum of 28,000 liters of insoluble substances in the tank, and if formation of lower density hydrates is taken into account, the centrifuged volume of sludge (assuming 20% supernate remaining in the centrifuged material) would be about 57,000 liters.

A list[1] of certain specified constituents in the tank with their allowable limits[7] is shown in Table 2.4. Columns A, B, and C in Table 2.4 are obtained from the records kept at West Valley. This is claimed to be the most accurate information available in West Valley,[8] because these data are obtained from what went into the tank which has never leaked. Table 2.4 also shows the results of the supernate sample analysis. There is a considerable amount of discrepancy in the data given in Column C and the results of the supernate analysis. The differences between the numbers have been recognized but not understood. This is of particular importance and concern because the content must be known more precisely before a specific waste solidification process can be used. Some examples of these discrepancies are: (i) Na present in the supernate is 11 x 10^6 moles (sample analysis) but 15 x 10^6 moles are expected to be present in the tank. The difference between these two numbers can be explained if the rest is present in the sludge, but this cannot be ascertained. After reviewing the available data, Reference 1 concluded that 1.28 x 10^7 moles or 1.09 x 10^6 kg of sodium nitrate is present in the tank. The uncertainty in the estimate of the constituent is then 20%. (ii) The amount of Cl present in the tank should be only 8.4 x 10^3 moles, but the supernate sample analysis gives a figure of 111.6 x 10^3 moles, which is an order of magnitude higher.

Since there are no routine analyses of fission products done (except for cesium), radioactive inventory estimates were made in Reference 1 by assuming reasonable average data of the reactor fuel received at West Valley. The code ORIGEN[9] was used to generate these data. These are shown in Table 2.5. The inventory includes curies of isotopes present and some of their heating rates. There is a large discrepancy between these numbers and the numbers given in Reference 5 (NUREG-0043). The difference has been largely attributed to the Cs-137 activity which was the basis of the radioactive inventory in Reference 4. The older estimate[10] of 18 million curies of Cs-137 activity (used in Reference 4 estimates) has been revised to be 11.9 million curies. The calculated inventory of the actinide elements is shown in Table 2.6. The Am-241 activity in this Table, however, does not take into account the buildup from Pu-241 decay.

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Table 2.4

Contents of Waste Tank 8D2 in West Valley From[1]
NFS Waste Surveillance Program Quarterly Report to New York State ERDA[a]

Component	A Moles Beginning of Dec. (x 10)	B Moles Added During Dec. (x 10)	C Moles at End of Dec. (x 10)	D Concentration at End of Dec. M	E Supernate Analysis Molarity	E Supernate Analysis Moles in Supernate (x 10)	Maximum Allowable Limit[7] M
Na + K	15,286.2	0.5	15,286.7	7.20	5.4	11,241.[f]	8.0
Fe	660.2	0	660.2	0.31	0.0045	8.3	0.6
					<0.0043 (Ni)	<8.3	
Cr + Ni	83.7	0	83.7	0.04	0.01 (Cr)	20.8	0.2
F	5.6	0	5.6	<0.01	---	---	0.2
Mo	12.0	0	12.0	0.01	0.02	4.29	0.1
Al	227.8	0	227.8	0.11	<0.0093	<18.7	0.4
P	188.5	0	188.5	0.09	<0.008	<16.7	0.2
C O	172.9	0	172.9	0.08	<0.05	<104.1	0.1
Cl	8.4	0	8.4	<0.01	0.054 (1.9 mg/mL)	111.6	0.1
So	640.2	0	640.2	0.30	0.20 (19 mg/mL)	412.0	1.0
OH(excess)	385.9	0	385.9	0.18	0.236	491.3	---

[a]Modified from Report for Fourth Quarter 1977. Report No. 9-78-09, dated January 17, 1978.
[b]These are calculated values, based on reported moles at end of period and volume.
[c]Based on laboratory analysis of quarterly supernate sample. These analyses are reported to New York State but are not used to calculate the molarities shown in Column D.
[d]Calculations assume 550,000 gallons (2.1×10^6 L) of supernate.
[e]Assumed to be 99.9% Na.
[f]The sodium analysis reported here is based on emission spectroscopy and is believed (by NFS) to be accurate to ±20%. The true value of sodium content is probably greater than that reflected by the supernate analysis but less than the Column A figure.
[g]NFS reports that the emission spectroscopy analysis for molybdenum yields results with a relatively narrow error band.
[h]NFS states that a maximum of approximately 11 tons of aluminum nitrate (as the nonahydrate) has been purchased over the lifetime of the plant. This is equivalent to 29,400 moles of aluminum, as Al; the lower figure has been used in subsequent calculations.
[i]It is very doubtful that there is any oxalate present, based on review of analyses and NFS statements regarding processes involving oxalate. The oxalate was disregarded in subsequent calculations.

Table 2.5

Calculated Radioactivity Inventory and Decay Heating Rates[a] in the Neutralized Tank at West Valley[1]

Fission Product Isotope	Half Life	Percent Assumed In Sludge	Curies At Time of Discharge	Curies (1978) In Sludge	Curies (1978) In Supernate	Curies (1978) Total	Curies (1985)	Watts At Time of Discharge	Watts 1978	Watts 1985
Se-79	6.50 E4 y	0.0	---	---	5.0 E1	5.0 E1	---	---	---	---
Sr-90	2.90 E1 y	99.0	1.2 E7	8.2 E6	8.3 E4	8.3 E6	7.6 E6	1.42 E4	9.80 E3	8.97 E3
Y-90	6.40 E1 h	99.9	1.2 E7	8.2 E6	8.3 E4	8.3 E6	7.6 E6	6.71 E4	4.64 E4	4.25 E4
Zr-93	1.50 E6 y	99.9	2.5 E2	2.5 E2	1	2.5 E2	2.5 E2	2	2	2
Nb-93m	1.36 E1 y	99.9	---	1.1 E2	1	2.0 E1	---	---	---	---
Tc-99	2.13 E5 y	0.0	1.9 E3	---	1.9 E3	1.9 E3	1.9 E3	3	3	3
Ru-106	3.68 E2 d	99.0	6.0 E7	5.4 E4	5.5 E2	5.5 E4	5.5 E2	3.30 E4	3.00 E1	1
Rh-106	3.00 E1 s	99.0	6.0 E7	5.4 E4	5.5 E2	5.5 E4	5.5 E2	1.30 E6	1.16 E3	1.20 E1
Pd-107	6.50 E6 y	99.0	---	6.0	1	6.0	---	---	---	---
Sb-125	2.71 y	99.0	7.9 E5	6.0 E4	6.0 E2	6.1 E4	1.1 E4	6.60 E3	5.10 E2	1.00 E2
Te-125m	5.80 E1 d	0	---	---	2.5 E4	2.5 E4	---	---	---	---
Sn-126	1.05 E5 y	99.0	---	4.0 E1	1	4.0 E1	---	---	---	---
Sb-126	1.90 E1 m	99.0	---	4.0 E1	1	4.0 E1	---	---	---	---
Sb-126m	1.25 E1 d	99.0	---	4.0 E1	1	4.0 E1	---	---	---	---
Cs-134	2.06 y	0	1.3 E7	---	4.3 E5	4.3 E5	3.9 E4	1.30 E5	4.70 E3	3.00 E1
Cs-135	3.00 E6 y	0	---	---	3.5 E1	3.5 E1	---	---	---	---
Cs-137	3.02 E1 y	0	1.4 E7	---	1.1 E7	1.1 E7	9.0 E6	1.70 E4	1.34 E4	1.11 E4
Ba-137m	2.50 m	0	1.3 E7	---	1.0 E7	8.4 E6	8.4 E6	5.05 E4	3.61 E4	3.03 E4
Ce-144	2.84 E2 d	99.9	3.2 E8	3.3 E4	3.0 E1	3.3 E4	3.3 E2	2.00 E5	2.00 E1	1
Pr-144	1.70 E1 m	99.9	3.2 E8	3.3 E4	3.0 E1	3.3 E4	3.3 E2	2.40 E6	2.40 E2	2
Pm-147	2.62 y	99.9	1.5 E7	6.5 E5	6.6 E2	6.6 E5	1.1 E5	5.90 E3	2.60 E2	4.50 E1
Sm-151	9.30 E1 y	99.9	2.3 E5	2.0 E5	2.1 E2	2.1 E5	2.0 E5	1.00 E2	9.60 E1	9.40 E1
Eu-152	1.34 E2 y	99.9	1.1 E3	6.6 E2	1	6.6 E2	4.5 E2	6	4	3
Eu-154	8.20 y	99.9	6.6 E5	2.8 E5	3.0 E3	2.8 E5	1.5 E5	1.06 E4	4.48 E3	2.40 E3
Actinides		99.9	---	8.5 E4	8.0 E1	8.5 E4	6.8 E4	---	1.2 E3	1.1 E3
TOTAL				1.8 E7	2.1 E7	3.9 E7		4.20 E6	1.17 E5	9.55 E4

[a]Long lived isotopes producing less than one watt are excluded.

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Table 2.6

Calculated Actinide Element Inventory[a] in Tank 8D2 at West Valley[1]

Isotope	Half-Life	Kilograms		Curies		Watts	
		1978	1985	1978	1985	1978	1985
Np-237	2.14 x 10^6 y	32	32	23	23	0.7	0.7
Np-239[b]	2.35 d	---	---	220	220	4	4
Pu-238	8.87 y	0.07	0.07	1660	1660	56	56
Pu-239	24370 y	29	29	1800	1800	56	56
Pu-240	6580 y	4.9	4.9	970	970	30	30
Pu-241	13.2 y	1	1	48300	34500	6	6
Pu-242	3.79 x 10 y	0.05	0.05	0.7	0.7	<1	<1
Am-241	458 y	5.8	5.8	18800	18600	620	620
Am-242	16.01 h	2 x 10	2 x 10	190	190	1	1
Am-242m	152 y	0.02	0.02	190	190	7	7
Am-243	7950 y	1.2	1.2	220	220	7	7
Cm-242	162.5 d	<<<1	---	---	---	<1	<1
Cm-244	17.6 y	0.15	0.11	12500	9200	430	320
Cm-245	14000 y	0.01	0.01	1	1	<1	<1
Cm-246	6600 y	0.001	0.001	0.2	0.2	<1	<1
Totals		72.2	74.1	84800	67600	1210	1100

[a]Uranium not included. It is estimated that approximately 2500 kg of unrecovered uranium from the Purex process are in the tank.
[b]In equilibrium with the Am-243 parent.

The total amount of plutonium present in the tank is 35 kg and was determined from chemical analysis of all solution transferred to the tank. The accuracy[1] of the result is believed to be ±6.1%. The calculated value of neptunium may be high because some of the neptunium may have been extracted away along with uranium in the Purex process.

Finally, Table 2.7 gives a list of the estimated[1] composition of the neutralized waste in West Valley. Several assumptions concerning the chemical nature of the various elements present were made in obtaining this list. For example, the tank contains carbonates but this has not been taken into account in this list. Similarly, it is assumed that no plutonium exists in the supernate. On a mass basis, these assumptions will have an insignificant effect, but must be considered in the final disposal of the waste. Table 2.7 shows that there are about 100 MT of sludge solids (anhydrous) and about 1000 MT of soluble salts in the neutralized waste tank in West Valley. The density of solids in the supernate varies[1] from 540 to 570 g/L.

Table 2.7

Estimated Composition of Contents of Waste Tank 8D2[a] at West Valley[1]

Compound	In Sludge Moles ($\times 10^3$)	In Sludge Kilograms	In Solution Moles ($\times 10^3$)	In Solution Kilograms
$(Na,K)_2 SO_4$	---	---	630	89800
$(Na,K) NO_3/NO_2$	---	---	12300	1090000
(Na,K) OH	---	---	386	15400
(Na,K) Cl	---	---	8.4	500
$Fe(OH)_3$	460	49500	---	---
$FePO_4$	190	28500	8.3	1250
$Cr(OH)_3$[b]	38	3900	21	2100
$Ni(OH)_2$[b]	18	1700	<8.3	<750
$Al(OH)_3$	11	830	---	---
AlF_3			<18.7	<1570
Rare earth $[M(OH)_3]$[c]	8.3	1620	---	---
Fission products,[f] $(M'SO_4)$	5.5	1240	2.9[d]	550
$[M'(OH)_4]$	9.6	1580	---	---
(M_2O_3)			---	---
(MO_{x-1})	0.7	210	1.2[e]	220
Mo, as Na phosphomolybdate $(Na_3PO_4\text{-}12MMoO_3)$	7.8	980	4.2	500
Mn, as MnO_2(est'd)	25	2200	---	---
Pu, as PuO_2	---	35	---	---
U, as $Na_2U_2O_7$	21	6700	---	---
Actinides, total as oxides	---	19	---	---
TOTALS		99000		1203000
		~280000 liters @ assumed dry density of 3.5 g/cc		557 g/L

[a]Based on composite of information from NFS (J. P. Duckworth, private communication), the NFS Waste Surveillance Report No. 9-78-09; and calculations based on total MWD burnup in fuel processed at NFS.

[b]Assuming ratio of chromium to nickel to be 19:9 (the ratio in which it occurs in 304 stainless steel).

[c]Average atomic weight - 143.9.

[d](Rb,Cs)NO_3, avg MW 189.52.

[e]Technicium as $NaTcO_4$. It is considered very doubtful that this is present; the Tc is most likely to be present as a Ba pertechnetate, coprecipitated with $BaSO_4$ (Martin & Miles, "Chemical Processing of Nuclear Fuels," Academic Press, 1958; p. 77).

[f]M is used as a common symbol for the cation. It could be any one of the fission products.

2.1.2 Acidic Waste in Tank 8D4

Total volume of waste	45,000 liters
Specific gravity	1.8
Temperature	40-50°C.

No sludge or precipitate are believed to be present in this waste.

The estimated[1] radioactive inventory and chemical composition of the waste in Tank 8D4 are shown in Table 2.8. These data are based on the NFS Safety Analysis Report. The code ORIGEN was used in obtaining the radioactivity values of these results. The values presented in NUREG-0043 agree with these data, but the basis of the two calculations differ considerably. Molar concentrations of different elements in the 8D4 Tank are also obtained from the NFS monthly records.[7] These are shown in Table 2.9.

Oak Ridge National Laboratory performed an analysis of a sample of this waste. ORNL results[1] for this Tank are shown in Table 2.10. There are again discrepancies between the calculated and measured values in Tables 2.8 and 2.10. An example of the discrepancy is given here. ORNL sample analysis of the acidic waste at West Valley indicates that 10,300 kg of Th (concentration 228 mg/ml) are present in the Tank (8D4). However, the fuel receipts record indicates that about 15,000 kg of Th should be present at West Valley. This may simply indicate that contrary to expectations, some acid waste has possibly precipitated at the bottom of the Tank.[3] It is learned[8] that the ORNL analysis was done for radioactivity analysis of the sample and not specifically for its chemical content. Also, they did not analyze the uranium and plutonium contents of the sample. Finally, the results are based on only one 50-ml sample and may not be representative of the Tank.[8]

This concludes the present status of HLW inventory at West Valley. It is felt that the data presented here are as good as can be obtained today from published sources.

2.2 DOE High Level Radioactive Waste

2.2.1 High Level Waste at Idaho National Engineering Laboratory

This inventory of the high level radioactive waste at Idaho National Engineering Laboratory (INEL) has been gathered from various unclassified documents and from a meeting with Idaho Chemical Processing Plant (ICPP) personnel. The literature surveyed indicates that the inventory is well documented.

To put the INEL waste in perspective, we first cite a GAO document[11] which states that only 3% of the AEC generated high level waste volume is stored at Idaho as compared to 25% and 72% stored, respectively, at the Savannah River Plant and at Hanford. The high level liquid waste at Idaho is generated from reprocessing of spent nuclear fuel at the Idaho Chemical Processing Plant. The spent fuel came primarily from the naval propulsion reactors and from various experimental or test reactors. The high level liquid wastes are stored in large doubly contained underground stainless steel tanks for a period not to exceed

Table 2.8

Estimated Constitution of Thorex Waste in Tank 8D4[a] at West Valley[1]

Radioactive Inventory	Ci
Thorium	3.55
Uranium	~0.01
Fission products (includes cesium, 7.47 x 10^5 Ci, and Strontium, 4.83 x 10^5 Ci)	2.43 x 10^6

Chemical Composition	MT
$Th(NO_3)_4$	32
$Al(NO_3)_3$	3.4
NaF	0.2
HNO_3	3.0
HPO_4	0.2
H_3BO_3	0.4
$M(NO_3)_x$ (M = Fe, Cr, Ni)	0.8

[a]Based on date in NFS Safety Analysis Report.

Table 2.9

High Level Liquid Waste in the Acidic Tank at West Valley From Waste Surveillance Program[7]

Component	Moles at End of April 1973 (x10^3)	Molar Concentration (April/December 1973)	Maximum Allowable Moles
Th	66.81	1.46	2.3
Al	16.16	0.35	0.5
F	4.70	0.10	0.2
HNO_3	46.79	1.03	2.0
P	2.08	0.04	0.2
Cl	0.0	0.0	0.05

Table 2.10

ORNL Analysis of Acidic Waste at West Valley[1]
(September 23, 1978)

Element	Concentration	Total in Tank 8D4[a] (kg)
Fe (mg/mL)	48	2,200
Cr "	12	540
Ni "	7	320
K "	3.4	150
Al "	10.3	460
Cl "	1.4	63
S "	0.7	32
Mn "	0.7	32
B "	0.4	18
Ca "	0.15	6.8
Na "	6.9	310
Th "	228	10,300
Sm (μg/mL)	200	9
Nd "	1000	45
Pm "	<30	1.4
Ce "	350	16
Pr "	180	8.1
La "	180	8.1
Cs "	700	32
Ba "	500	23
Pd "	100	4.5
Rh "	120	5.4
Ru "	350	16
Mo "	550	25
Tc "	180	8.1
Zr "	120	5.4
Sr "	180	8.1
Y "	80	3.6
Rb "	80	3.6
Se "	60	2.7
As "	<10	<0.5
Zn "	180	3.6
Cu "	60	2.7
Co "	30	1.4
V "	<20	0.9
Ti "	<150	6.8
P "	220	9.9
F "	20	0.9

[a]Assuming 45,000 L of waste, representative sample.

Table 2.10 Continued

Radioactive Analysis

Isotope	Activity (dps/mL)	Extrapolated Activity in Tank (Ci)
Co-60	4.01×10^6	4.88×10^3
Cs-134	9.13×10^6	1.11×10^4
Cs-137	6.96×10^8	8.47×10^5
Er-154	$(7.4 \pm 0.8) \times 10^6$	9.00×10^3
Sr-90	6.59×10^8	8.02×10^5

five years. The liquid waste is then transferred to the Waste Calcining Facility (WCF), where it is solidified and stored in a retrievable form in doubly contained underground bins.[12] There are some other high level wastes stored at Idaho. These will be briefly mentioned in the discussion of spent fuel in Section 2.2.1.3 below.[13,14]

2.2.1.1 High Level Radioactive Liquid Wastes (HLLW)

The tank farm for the liquid wastes consists of 18 stainless steel tanks, nine of which are of 300,000-gallon capacity and two of 318,000-gallon capacity. The rest are of 30,000-gallon capacity. There are five additional 18,000-gallon tanks used for temporary waste storage.[15] Four of the 30,000-gallon tanks, WM-103 to WM-106, were employed in the past to store wastes from early processes used at the ICPP. All of these tanks are out of service at present because they do not meet the secondary containment criterion for storage of HLLW. The stainless steel sulfate wastes in these tanks have been blended with the fluoride waste. The present tank farm liquid waste inventory is given in Table 2.11.[15] It should be mentioned that the volumes of liquids in the tanks change with time because of additions to the tanks and transfers from the tanks to the WCF. The accuracy of the liquid volume estimates for these tanks is 5% or better.

The 300,000-gallon tanks are well monitored. A level indicator in each tank is set to alarm when the volume reaches 285,000 gallons. Tank pressure is also measured and transmitted to the control room. Specific gravity of liquid waste is measured continuously and an alarm is activated at a specific gravity of 1.35 (the design limit is 1.40). The temperature of each tank is controlled below 50°C for corrosion control and is continuously recorded. There are 24 temperature points for each cooled tank and 10 for each uncooled tank. The eight 300,000-gallon cooled tanks each contain a 4.7-million Btu/hr cooling capacity system. This is done by surrounding 30 coils of stainless steel pipe around the interior walls and floors of the tanks. The water pressure in the cooling coils is greater than the pressure in the waste tank.

Table 2.11

Radioactive Liquid Waste Inventory in Different Tanks at ICPP[15]
(August 31, 1980)

Tank Number[a]	Waste Type	Volume, m^3
WM-182	High Level Zirconium Fluoride Waste	80.8
WM-185		713
WM-187		1038
WM-188		1068.5
WM-189		1073.0
WM-100	High Level Non-Fluoride Waste	48.5
WM-101		68.9
WM-102		66.2
WM-183	High Level Aluminum Nitrate Waste	1060
High Level Sub-Total		5216.9
WM-184[b]	Intermediate Level Waste	1046
WM-186[b]		1064
WM-180		926.4
WM-181[b]		1000.5
Intermediate Level Sub-Total		4036.9
Total Intermediate and High Level Liquid Wastes		9253.8

[a]Tank WM-190 is a spare tank for high level waste.
[b]These tanks do not have cooling coils.

The high level liquid waste is primarily the first cycle raffinate from the solvent extraction of dissolved fuel solutions. It contains about 99.9% of the fission products.[16] These are produced at the ICPP and consist of (1) acid aluminum waste from various test reactor fuels, (2) fluoride bearing wastes from zirconium and from zirconia-containing fuels, (3) a small amount of stainless steel sulfate from development reactor fuels such as organic moderated reactor experiments, (4) acid stainless steel nitrate waste from the electrolytic dissolution of EBR-II fuel, and (5) an acid waste from the recovery of uranium from nuclear rocket fuel. The first cycle raffinate waste is self heating in both the liquid and solid forms. The second and third cycle raffinates are of low heat but high level liquid wastes. Approximately 10,000 to 35,000 gallons of second and third cycle radioactive wastes are produced each year.[17] These are kept in the non-cooled waste tanks.

Chemical compositions of the liquid wastes are given in Tables 2.12 through 2.15.[15] As seen in these tables, the chemical composition of the high level liquid waste varies according to types. Acid aluminum waste is primarily aluminum nitrate and nitric acid. Fluoride waste consists of aluminum and zirconium fluorides and nitrates along with nitric acid. Average composition of aluminum and zirconium wastes are shown in Table 2.16.[16,17] The uncertainties of these estimates have mostly not been determined. The accuracy is better for the high concentration elements and could otherwise be in error by a factor of ±2.[15]

Radiochemical composition of the tank farm inventory is shown in Tables 2.17 through 2.20.[15] Further details of the actinides in the stored first cycle raffinate are shown in Table 2.21.[12,16] This table indicates that the ratio of Np-237 to Pu varies from about 5×10^{-3} to 10×10^{-3} in the first cycle raffinate. Table 2.21, however, mentions that the uncertainty of the experimental results is ±100%. Slansky[12] has estimated the concentrations of Am-241, Am-243, Cm-242, and Cm-244 for the average high level zirconium bearing waste to be 3.8×10^{-5}, 1.1×10^{-6}, 9.9×10^{-10} and 6.6×10^{-8} g/L, respectively. This is based on the isotopic content for Pu and U, and cooling times at the ICPP. These numbers differ by about 10% from those of Table 2.21. The decay heat generation rates in the tanks at ICPP are shown in Table 2.22.

To summarize, there is about 2.4 million gallons of high liquid waste stored in the tanks at INEL. The exact volumes of this were as reported in different documents are: 8980 m^3 (=2.37 million gallons) in NUREG-0527[13]; 350×10^3 cu ft (=2.62 million gallons) in the IRG report[6] and 8700 m^3 (=2.3 million gallons) in ERDA 77-43.[16] The Savannah River Strategy Document[18] quotes this high level liquid volume to be 2.5 million gallons in 1978 and reduced to 1.2 million gallons in 1990. The total inventory of the high level radioactive wastes at Idaho is shown in Figure 2.2 as a function of time from 1964 to 1974. It also shows the total liquid volume generated at Idaho and the volume left after partial calcination. The total activity of the liquid waste is estimated to be 1.1×10^7 curies in the NUREG document.[13]

There has been no intentional or unintentional leakage of radioactive waste to the ground at INEL since operation began in 1951.[13]

2.2.1.2 High Level Calcined Solid Waste

The high level liquid waste at Idaho is converted to granular solids by calcination to reduce the volume[a] (about 8:1 ratio)[15] and the cost[12] of storing huge volumes of liquid waste. The Waste Calcination Facility (WCF) at the ICPP has been solidifying high level wastes in a fluidized bed calciner since December 1963. As of October 1, 1979, 3.6 million gallons (13,630 m^3) of high level liquid waste had been calcined to 1978 m^3 of calcined waste.[19] Since its start, 5570 m^3 of high level aluminum waste, 6760 m^3 of high level zirconium fluoride waste, 1290 m^3 of sodium-bearing waste and 19 m^3 of stainless steel sulfate waste have been calcined. Table 2.23 gives a detailed description of the total calcine inventory by bin set and calcine type. The

[a]The volume reduction factor varies from operation to operation and ranges from 7 to 14.

Table 2.12

Chemical Composition of High Level Zirconium Fluoride Wastes at ICPP[15]
(August 31, 1980)

Constituent	Tank Number				
	WM-182	WM-185	WM-187	WM-188	WM-189
H^+ (N[a])	1.20	1.39	1.37	1.35	1.59
Al (M)	0.41	0.62	0.66	0.59	0.45
Zr (M)	0.19	0.34	0.43	0.30	0.23
B (M)	0.096	0.18	0.18	0.17	0.11
Fe (M)	0.007	0.006	0.007	0.01	0.009
Na (M)	0.05	----	0.03	----	0.10
F^- (M)	1.28	2.22	2.67	1.94	1.60
NO_3^-	2.58	2.51	2.57	2.63	2.60
Cl (g/mL)	----	----	----	----	170
K (M)	----	----	----	----	0.018
NP-237 (mg/L)	----	----	----	----	----
U (mg/L)	----	1.77	0.25	----	----
Pu (mg/L)	----	----	----	----	----
Undissolved Solids (g/L)	14	----	----	17	11

Table 2.13

Chemical Composition of High Level Non-Fluoride Wastes at ICPP[15]
(January 19, 1978)

Constituent, (M)	Tank Number		
	WM-100	WM-101	WM-102
H^+	0.66	1.26	1.05
Al	1.20	0.51	0.65
Zr	0.0083	----	----
Fe	0.073	0.64	0.064
Cr	0.0035	0.015	0.012
Ni	0.0017	0.0078	0.0058
F^-	0.0256	0.0068	0.0026
Cl^-	0.0008	0.0008	0.00085
Na	0.0255	0.0222	0.022
NH_4^-	0.51	0.51	0.5
NO_3^-	5.25	3.34	3.68
SO_4^{-2}	0.041	0.041	0.082
Mo	0.0008	0.0025	0.023
Rh	0.0002	0.00013	0.00012
Ru	0.0033	0.00087	0.00080

Table 2.13 Continued

Constituent, (M)	Tank Number WM-100	WM-101	WM-102
Gd	----	0.021	0.017
Pd	0.0004	----	----
Np-237 (mg/L)	----	----	0.08
U(mg/L)	7.9	----	2.0
Pu(mg/L)	4.3	----	11.9

Table 2.14

Chemical Composition of High Level Aluminum Nitrate Waste at ICPP[15] (December, 1977)

Constituent, (M)	Tank Number WM-183
H^+	1.4
Al	0.66
Fe	0.037
Cr	0.0081
Ni	0.0034
F^-	0.019
Cl^-	0.0070
Na	0.67
B	0.0026
Ca	0.075
K	0.10
NO_3^-	4.42
SO_4^{-2}	0.084
PO_4^{-3}	0.00085
Mg	0.030
Mn	0.0089
Cu	0.00079
Np-237 (mg/L)	----
U (mg/L)	----
Pu (mg/L)	----

Table 2.15

Chemical Composition of Intermediate Level Wastes at ICPP[15]

Constituent (M)	Tank Number			
	WM-180	WM-181	WM-184	WM-186
H^+	1.22	1.30	0.14	1.63
Al	0.55	0.4	0.59	0.37
Zr	<0.05	----	----	0.00011
Fe	0.06	1 (g/L)	0.00011	0.0217
F^-	0.22	0.04	0.018	0.0065
Cl^-	0.993 (g/L)	0.8 (g/L)	0.0454	0.0310
Na	1.26	1.6	2.44	1.59
Hg	----	0.3 (g/L)	0.00073	0.0028
B	0.019	0.2 (g/L)	0.0046	0.0148
K	0.20	0.21	0.128	0.225
NO_3^-	4.11	4.3	4.64	5.02
SO_4^{-2}	----	5 (g/L)	0.063	0.059
PO_4^{-3}	----	0.4 (g/L)	0.0277	0.0099
Mn	----	1 (g/L)	0.0046	0.0228
NP-237 (mg/L)	----	----	----	----
U (mg/L)	----	32	----	----
Pu (mg/L)	----	----	----	----
Undissolved solids (g/L)	16	----	----	----

Table 2.16

Average Composition of High Level Liquid Waste at the ICPP[15,16]

Component	Composition, Wt (%)	
	Aluminum Waste	Zirconium Waste
$Al(NO_3)_3$	24.1	---
AlF_3	---	---
ZrF_4	---	4.1
$NaNO_3$	0.3	0.02
$Hg(NO_3)_2$	0.3	---
$Ca(NO_3)_2$	---	13.8
H_3BO_3	0.1	0.9
HNO_3	4.5	6.6
H_2O	70.7	70.2
	100.0	100.0
Total Activity:	24 Ci/gal	3 Ci/gal

Table 2.17

Radiochemical Analysis of High Level Zirconium Fluoride Wastes at ICPP[15]

Radionuclide	Activity (Ci/m^3) WM-182 (1972)	WM-185 (1980)	WM-187 (1980)	WM-188	WM-189
Co-60	----	1	1.3	----	----
Sr-90	375	689	not available	780	400
Ru-106	2	21	35	6	2
Sb-125	3	13	24	9	3
Cs-134	16	107	154	44	14
Cs-137	445	749	1622	1000	440
Ce-144	3.5	125	184	23	4
Eu-154	4.5	11	22	9	3
Eu-155	3	5	12	5	2

Table 2.18

Radiochemical Analysis of High Level Non-Fluoride Wastes at ICPP[15]

Radionuclide	Activity (dps/mL) WM-100	WM-101 (7/5/78)	WM-102 (7/5/78)
H-3	----	8.26×10^3	5.21×10^3
Mn-54	----	5.88×10^5	1.92×10^5
Co-60	3.58×10^3	2.14×10^5	1.22×10^5
Sr-90	8.05×10^6	1.23×10^7	7.10×10^6
Nb-95	----	4.45×10^4	4.38×10^3
Zr-95	----	3.65×10^4	----
Ru-106	----	1.08×10^6	5.74×10^5
Sb-125	----	2.58×10^5	1.26×10^5
Cs-134	----	2.94×10^5	1.01×10^5
Cs-137	----	1.36×10^7	8.86×10^6
Ce-144	----	1.59×10^7	5.24×10^6
Eu-154	----	----	5.97×10^3
Eu-155	----	3.26×10^5	1.96×10^5

Table 2.19

Radiochemical Analysis of High Level Aluminum Nitrate Waste at ICPP[15]

Radionuclide	Activity (dps/mL) (1976) Tank No. WM-183
Co-60	----
Sr-90	----
Ru-106	5.7×10^5
Sb-125	----
Cs-134	1.3×10^5
Cs-137	7.7×10^6
Ce-144	6.5×10^6
Eu-154	2.1×10^4
Eu-155	----

Table 2.20

Radiochemical Analysis of Intermediate Level Wastes at ICPP Tanks

Radionuclide	Activity (dps/mL) WM-180[12]	WM-181	WM-184[15] (August 1972)	WM-186[15] (August 1972)
H-3	1.26×10^3	----	3.73×10^3	2.26×10^3
Sr-90	7.77×10^6	----	4.38×10^5	1.74×10^6
Nb-95	3.70×10^3	----	1.57×10^2	1.23×10^4
Zr-95	1.22×10^4	----	3.81×10^2	2.30×10^4
Ru-106	1.22×10^5	n.a.	5.52×10^3	3.18×10^5
Sb-125	1.11×10^5	----	8.77×10^3	2.47×10^4
Cs-134	3.7×10^4	----	9.61×10^2	3.87×10^4
Cs-137	8.88×10^6	----	4.49×10^5	2.09×10^6
Ce-144	2.81×10^4	----	1.49×10^3	1.03×10^5
Eu-154	2.59×10^4	----	1.07×10^3	9.71×10^3
Co-60			3.71×10^2	
Gross beta	2.85×10^7	----	1.61×10^6	7.70×10^6
Total U,[12] g/L	2.31×10^{-2}	----	2.01×10^{-2}	0.191
Total Pu,[12] g/L	9.85×10^{-4}	----	1.98×10^{-3}	----

Table 2.21

Actinide Analyses of Stored ICPP First Cycle Raffinate[a,b] [12,16]

Isotope	Tank Number WM-185 (g/L)	WM-187 (g/L)	WM-188 (g/L)
Total U	4.7 E-4	9.3 E-4	2.8 E-3
Np-237	1.19 E-5	1.57 E-5	0.97 E-5
Pu-238	4.7 E-4	2.7 E-4	3.7 E-4
Pu-239	1.39 E-3	0.80 E-3	1.08 E-3
Pu-240	3.40 E-4	2.44 E-4	2.91 E-4
Pu-241	1.51 E-4	1.09 E-4	1.26 E-4
Pu-242	4.78 E-5	4.05 E-5	4.16 E-5
Am-241	4.4 E-5	2.7 E-5	3.5 E-5
Am-243	1.2 E-6	0.78 E-6	0.95 E-6
Cm-244	7.3 E-7	4.5 E-7	5.8 E-7

[a]Pu-238, americium, and curium values are calculated.
[b]Experimental values have an estimated accuracy of ± 100%.

Table 2.22

Decay Heat Generation Rates in ICPP Tanks (September 1978)

Tank Number	Heating Rate (W/m^3)
High Level Zirconium-Fluoride Wastes	
WM-182	not available
WM-185	11
WM-187	23
WM-188	not available
WM-189	not available
High Level Non-Fluoride Wastes	
WM-100	not available
WM-101	7.35
WM-102	3.6
High Level Aluminum Nitrate Wastes	
WM-183	not available
Intermediate Level Wastes	
WM-181	not available
WM-180	not available
WM-186	0.51

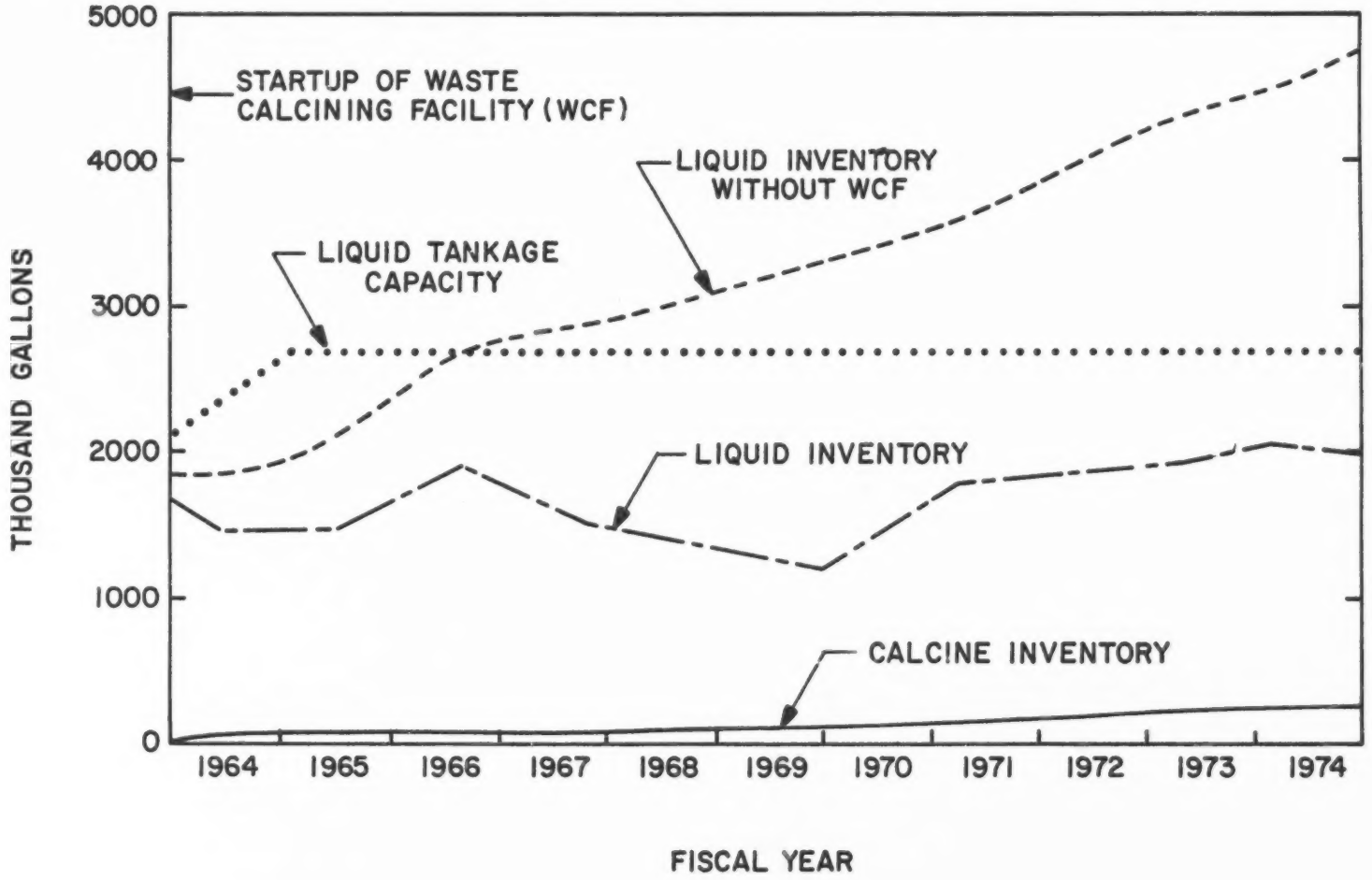

Figure 2.2. Idaho high level waste inventory from 1964-1974 (from GAO Document No. RED-75-309[11]).

Table 2.23

Inventory of Calcine in Storage at the ICPP[19]
As of October 1, 1979

Bin Set	Cold Alumina and Dolomite (Startup)	Alumina	Zirconium Fluoride	Zirconium-Sodium Blend (3.5:1)	Zirconium-Sodium Blend (5:1)	Stainless Steel Sulfate	Total Calcine Volume	Total Radionuclide Activity in Bin (10^5 Curies)
1(WC-115)	4.8	212.4	0.0	0.0	0.0	0.0	217.2	36.1
2(WC-136)	25.9	288.6	548.6	0.0	0.0	0.0	863.1	78.0
3(WC-140)	112.1*	63.9	548.0	110.1	62.2	1.4	897.7	69.6
TOTALS	142.8	564.9	1096.6	110.1	62.2	1.4	1978.0	183.7

*Note: The third bin set also contains about 6.0 m^3 of fluorapatite from calciner startup.

Savannah River Strategy Document[18] estimated the calcined volume to be 4500 m^3 in 1990. They have also suggested that this volume could go as high as 8500 m^3.

Typical compositions of raw feed solution to the waste calciner are given in Table 2.24.[19] The undissolved and dissolved solids in the liquid solution are made up of 99 weight percent nonradioactive material and about 1 weight percent radioactive waste. Liquid feed to the calciner also contains 15 to 20 volume percent of recycled off-gas scrubbing solution. Typical physical, chemical, and radiochemical properties of calcined solids from acid aluminum waste and fluoride bearing zirconium waste are shown in Table 2.25, constructed from three earlier references. The solidified waste granules are spherical and range from a fine powder to sand (<.01 mm to 0.7 mm). The chemical compositions of solid calcines vary according to type. Aluminum waste calcines are predominantly Al_2O_3 (88%) with small amounts of Na_2O, N_2O_5 and HgO. Zirconium waste calcines are mixtures of CaF_2 (54%), and ZrO_2 and Al_2O_3. The principal radioisotopes contained in the granules are Sr-90, Cs-134, Cs-137, and Ce-144. Latest data published by ICPP[19] on the calcine composition are shown in Table 2.26. It contains new data on the sodium-bearing and stainless-steel-sulfate wastes. The aluminum and zirconium data are slightly different, but similar to those of Table 2.25. Table 2.27 gives a detailed radionuclide composition of average calcine wastes at INEL.[20] Table 2.28 gives the time dependence of the activity and the heating rates of the calcines from 1 to 100,000 years.[20] These time tables are obtained from the work of Sutherland and Bennett,[20] who used the ORIGEN code[21] to derive these numbers. These data are graphically shown in Figures 2.3 and 2.4 and compared with the glass wastes from the Savannah River plant and the Hanford Reservations. The INEL wastes in calcine form have the second highest heat generation rates for the first 200 years, but they surpassed the Hanford wastes. These are further discussed later.

Some radioactive calcines were retrieved after 10-12 years of storage and their physical, chemical, and radiochemical contents were examined.[22] It was found that their chemical and physical properties have remained similar to freshly produced calcines. Table 2.29 shows the results of analysis of the chemical compositions of alumina and zirconia calcines. There are some apparent discrepancies between Tables 2.25 and 2.29. Note that Table 2.25 lists constituents as oxides, and Table 2.29 gives results for element. Table 2.25 shows that in aluminum waste there is approximately 47% Al and 0.96-1.48% Na, which is consistent with the Table 2.29 footnote. Similarly, Table 2.25 gives 11.6% Al, 27.8% Ca, 26.4% F, and 15.8% Zr in Zr waste. The Al and Ca contents are slightly different from the Table 2.29 footnote, and reflect the possible larger range of the chemical compositions of the calcine waste. Thus, a variation of about 20% or more in the compositions may be expected. Tables 2.30 and 2.31 show the results of radiochemical analysis of retrieved alumina and zirconia calcine wastes.[22] The Np-237 content of the calcines as measured by Staples, et al.[22] and reported in Tables 2.30 and 2.31, is rather high compared to the total Pu present in the solid. The first raffinate waste analyses suggest that the Np-237-to-Pu ratio should be about 6×10^{-3}. It has been recently ascertained[23] that improved analyses of the retrieved calcine samples have been performed, and the Np-237 content has been updated to about 7 μg/g of calcine for the alumina calcine and about 3 μg/g of calcine for the zirconia

Table 2.24

Compositions Representative of ICPP Raw Waste[19]

Component, M	Aluminum Nitrate Type: Nitric Acid Type	Aluminum Nitrate Type: Ammonium Nitrate Type	Zirconium-Fluoride Type	Stainless Steel Sulfate Type	Sodium Bearing Type (Tank WM-180)
H^+	0.8	0.13 acid deficient	1.4	3.15	1.40
Al	1.5	1.65	0.68	----	0.56
Zr	----	----	0.41	----	----
Fe	0.005	0.004	0.0076	0.07(ppm)	0.0111
Na	0.065	0.1	----	----	1.59
F	----	----	3.24	----	0.00065
Hg	0.018	0.02	----	----	0.0050
NH_{4^+}	----	1.4	0.02	----	----
NO_{3^-}	5.4	7.1	2.28	2.48	4.38
Cr	----	----	0.015	0.02	----
B[a]	----	----	0.19	----	0.0057
Sn	----	----	0.0047	----	----
Ni	----	----	----	0.01	----
SO_4	----	----	----	0.56	0.035
K[a]	----	----	----	----	0.24
Cl^{-a}	----	----	----	----	0.032
PO_{4^-}	----	----	----	----	0.020
Mn^{4-}	----	----	----	----	0.023

[a]Components such as B and Ca that are not present in the raw wastes prior to calcination; e.g. B is added to the aluminum nitrate type waste to prevent the formation of alpha-alumina during calcination, calcium is added to zirconia waste to complex F also K and Cl may be present in trace amounts.

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Table 2.25

Typical Properties of Calcined Solids at ICPP

Physical Properties	Aluminum Waste (non-fluoride) Ref. 12	Ref. 14	Ref. 16	Zirconium Waste (fluoride) Ref. 12	Ref. 14	Ref. 16
Mass median particle diameter (mm)	0.56-0.70	0.56-0.70		0.6-0.8	0.6-0.8	
Bulk density g/cm^3	1.0-1.2	1.0-1.2	1.12	1.7	1.7	1.6
Heat generation rate Btu/hr-lb	0.3-0.8	0.06-0.16	0.4	0.1-0.2	0.02-0.04	0.1-0.2
Composition (wt %)						
Metallic oxides			89.2			43.3
Zirconium as ZrO_2				21.4	21.4	
Calcium as CaF_2				54.2	54.2	54.2
Aluminum as Al_2O_3	88.2-89.1	88.2-89.1		21.9	21.9	
Sodium as Na_2O	1.3-2.0	1.3-2.0				
Nitrogen as N_2O_5	3.9-4.1	3.9-4.1	4.0	1.9	1.9	<1.0
Mercury as HgO	2.9	2.9				
Water	2.0	2.0	<2.0	0.6	0.6	<1.0
Gross fission product oxides	0.6	0.6				
Fission product and other oxides or fluorides			4.8			0.5
Radioactive Properties						
Principal radioisotopes Ci/kg:						
Strontium-90	10.8-15.9	2.23-3.3[a]	13.2	1.8-5.5	0.36-1.14[a]	3.52
Cesium-134	0.44-0.88	0.09-0.18[a]				
Cesium-137	11.5-17.6	2.4-3.6[a]	17.6	2.2-6.6	0.45-1.4[a]	2.86
Cesium-144	2.9-38.1	0.6-7.9[a]				
Ruthenium-106	0.09-1.1	0.02-0.23[a]		0.02-0.07	0.004-0.01[a]	
Zirconium-Niobium-95	0-1.1	0.0-0.23[a]				
Promethium-147	7.3-23.4	1.5-4.8[a]				
Plutonium			0.02			0.01
Plutonium-238		0.002			0.01	
Plutonium-239		0.0004			0.0001	
Plutonium-240		0.0001			0.00006	

[a]It is suspected that there was an error in the conversion factor when these numbers were generated in Reference 4.

Table 2.26

Composition of ICPP Calcines[19]

Component	Type of Calcine			
	Alumina	Zirconia	Zr-Na (5:1 Blend)	Stainless Steel Sulfate
	Composition (wt %)			
Fe_2O_3				7.0
Al_2O_3	82-95	13-17	10-16	4.4[a]
Na_2O	1-3		6-8	
ZrO_2		21-17	16-19	
CaF_2		50-56	33-39	
$Al_2(SO_4)_3$				81.3[a]
CaO		2-4	13-17	
NO_3	5-9	0.5-2	7-9.5	
B_2O_3	0.5-2	3-4	2-3	
Cr_2O_3				2.0
NiO				0.9
Miscellaneous Fission Product and Actinides	0.5-1.5	0.5-1.5	0.5-1.5	4.4
	0.2-1	0.2-1	0.2-1	0.2-1

[a]Aluminum nitrate (2.2 M) was blended with the raw stainless steel sulfate waste before calcination.

Table 2.27

Radionuclide Content of INEL Defense High Level Waste in Calcine Form[20]

Isotope	Curies/m^3*	Isotope	Curies/m^3
Se-79	8.3 - 2	Pm-147	1.6 + 4
Rb-87	4.7 - 6	Sm-151	2.2 + 2
Sr-90	1.7 + 4	Eu-154	2.3 + 2
Y-90	1.7 + 4	U-233	1.6 - 9
Zr-93	4.0 - 1	U-234	5.6 - 7
Nb-93m	9.8 - 2	U-235	2.3 - 6
Tc-99	2.7 + 0	U-236	1.3 - 5
Ru-106	1.3 + 3	U-237	6.2 - 9
Rh-106	1.3 + 3	U-238	1.3 - 11
Pd-107	2.6 - 3	Np-237	6.2 - 5
Sn-126	4.2 - 2	Pu-238	9.1 + 1
Sb-126m	4.2 - 2	Pu-239	9.1 - 1
Sb-126	4.2 - 2	Pu-240	8.5 - 1
Cs-134	4.3 + 3	Pu-241	2.1 + 2

Table 2.27 Continued

Isotope	Curies/m^3*	Isotope	Curies/m^3
Cs-135	9.8 - 2	Pu-242	2.3 - 3
Cs-137	1.7 + 4	Am-241	1.3 + 0
Ba-137m	1.6 + 4	Am-243	1.1 - 2
Ce-144	1.1 + 4	Cm-242	8.5 - 1
Pr-144	1.1 + 4	Cm-244	6.8 - 1

*Ci/m^3 of the final product.

calcine. Staples[23] also suggested that the initial Np-237/Pu ratio in different fresh samples could be different because of different processings and mixings done during the processing, and this may explain the discrepancy.

Because the fuel processed at ICPP is low in U-238, the amount of actinides produced is small. The calcine contains[16] about 16 grams of total actinides per cubic meter (0.001 lb/ft^3). Decay curves for the important actinides in ICPP calcine from a typical raffinate are shown in Figure 2.5.

The total radionuclide activity of the calcines is 18.4×10^6 curies as of October 1, 1979. The distribution of the radioactivity among the three bins is given in Table 2.23 and 2.32. Table 2.32 shows that there are 8.5×10^6 Ci of Sr-90 and 9.7×10^6 Ci of Cs-137 in the calcined wastes in Idaho as of October 1, 1979. The total radioactivity between calcined solids and high level liquids at ICPP is estimated[17] to be 85×10^6 Ci in 1978 and could be somewhere between 300 and 500 million Ci in 1990. The plutonium content of Idaho waste has been estimated[24,17] to be 20 kg in 1977 and 50 kg in 1990.[25,17]

2.2.1.3 Spent Fuel and Other Radioactive Solid Wastes

There is an E-shaped spent fuel storage building with three storage basins which contain most fuels to be processed at the ICPP. Two of the basins provide storage for about 4,000 separate fuel elements. The third basin can store some 2,000 to 3,000 fuel elements.[14]

There are three other long term spent fuel storage areas at ICPP.[13] These are the Peach Bottom Fuel Storage Facility, the Graphite Fuel Storage Facility, and the Fermi I Fuel Storage Area. These fuels are in storage for many years. The number of fuel elements stored in these facilities are: 1603 Peach Bottom fuel elements, 4717 graphite Rover fuel elements and 510 Fermi I fuel elements.[23]

The Argonne National Laboratory-West has some HLW stored at the Radioactive Scrap and Waste Facility. The volume of the material is small[13] (less than 100 m^3), but the activity is high, about 9×10^6 Ci. Some of these are EBR-II irradiated blanket subassemblies. There is no firm plan for reprocessing this fuel.

Table 2.28

INEL Defense High Level Calcine Waste Curie Content and Heating Rates as Functions of Time[20]

Waste Age (yr)	Curies/m^3	Watts/m^3*	Waste Age (yr)	Curies/m^3	Watts/m^3
1	9.09 + 4	2.73 + 2	1000	7.70 + 0	1.09 - 1
2	7.98 + 4	2.33 + 2	2000	6.16 + 0	6.30 - 2
3	7.31 + 4	2.11 + 2	3000	5.81 + 0	5.17 - 2
4	6.85 + 4	1.97 + 2	4000	5.66 + 0	4.74 - 2
5	6.50 + 4	1.87 + 2	5000	5.57 + 0	4.47 - 2
6	6.22 + 4	1.79 + 2	6000	5.48 + 0	4.24 - 2
7	5.99 + 4	1.73 + 2	7000	5.40 + 0	4.03 - 2
8	5.78 + 4	1.67 + 2	8000	5.33 + 0	3.85 - 2
9	5.60 + 4	1.62 + 2	9000	5.27 + 0	3.67 - 2
10	5.43 + 4	1.58 + 2	10000	5.21 + 0	3.51 - 2
20	4.18 + 4	1.23 + 2	20000	4.75 + 0	2.41 - 2
30	3.29 + 4	9.69 + 1	30000	4.47 + 0	1.83 - 2
40	2.60 + 4	7.66 + 1	40000	4.27 + 0	1.49 - 2
50	2.05 + 4	6.06 + 1	50000	4.11 + 0	1.26 - 2
60	1.62 + 4	4.81 + 1	60000	3.97 + 0	1.11 - 2
70	1.28 + 4	3.81 + 1	70000	3.85 + 0	9.95 - 3
80	1.01 + 4	3.03 + 1	80000	3.75 + 0	9.14 - 3
90	7.98 + 3	2.42 + 1	90000	3.65 + 0	8.55 - 3
100	6.31 + 3	1.93 + 1	100000	3.56 + 0	8.11 - 3
200	6.52 + 2	2.50 + 0			
300	9.68 + 1	6.63 - 1			
400	3.04 + 1	3.44 - 1			
500	1.72 + 1	2.38 - 1			
600	1.24 + 1	1.85 - 1			
700	1.01 + 1	1.55 - 1			
800	8.90 + 0	1.35 - 1			
900	8.17 + 0	1.20 - 1			

*Waste age here refers to years from calcination.

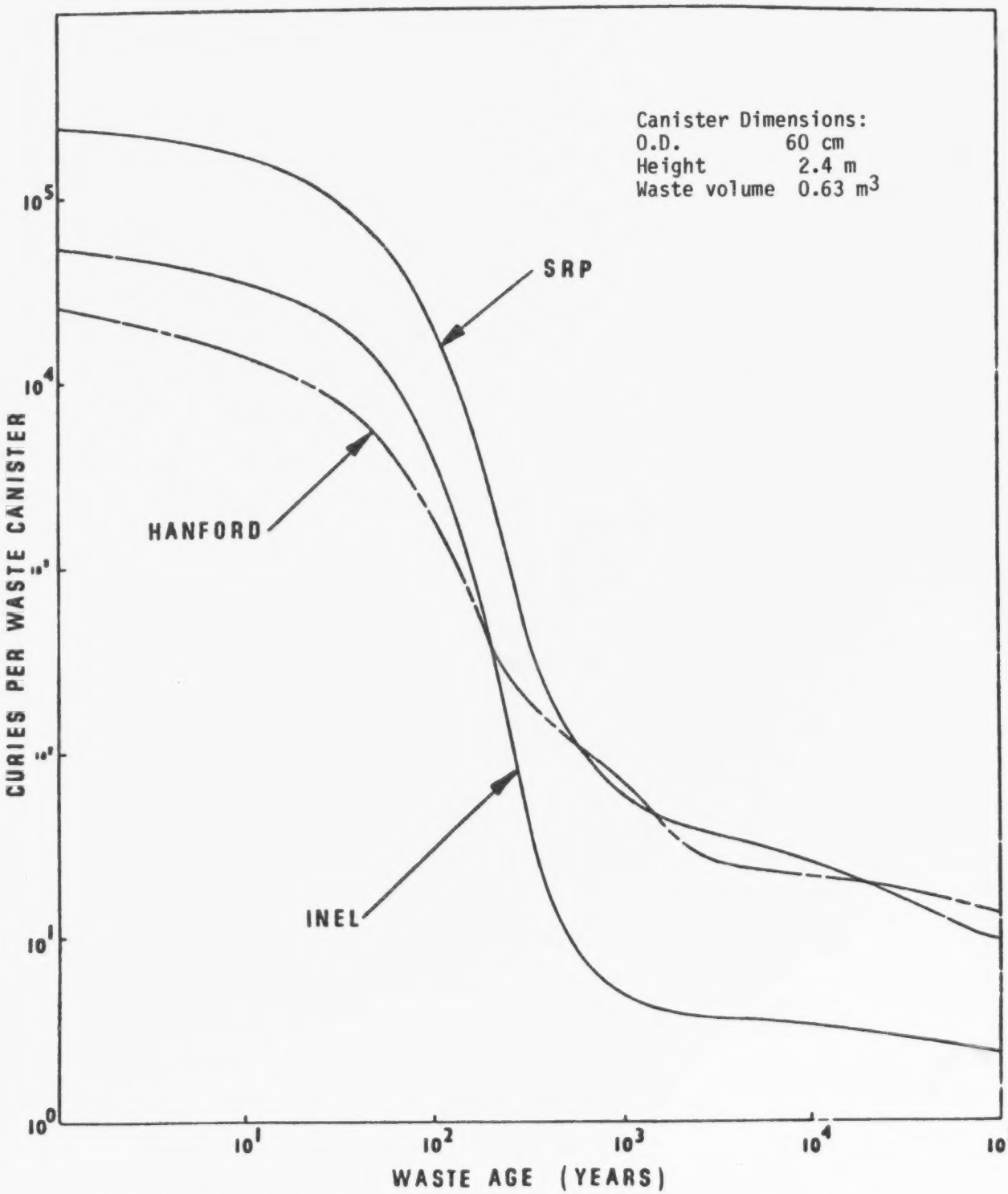

Figure 2.3. High level defense waste total curie content.[20]

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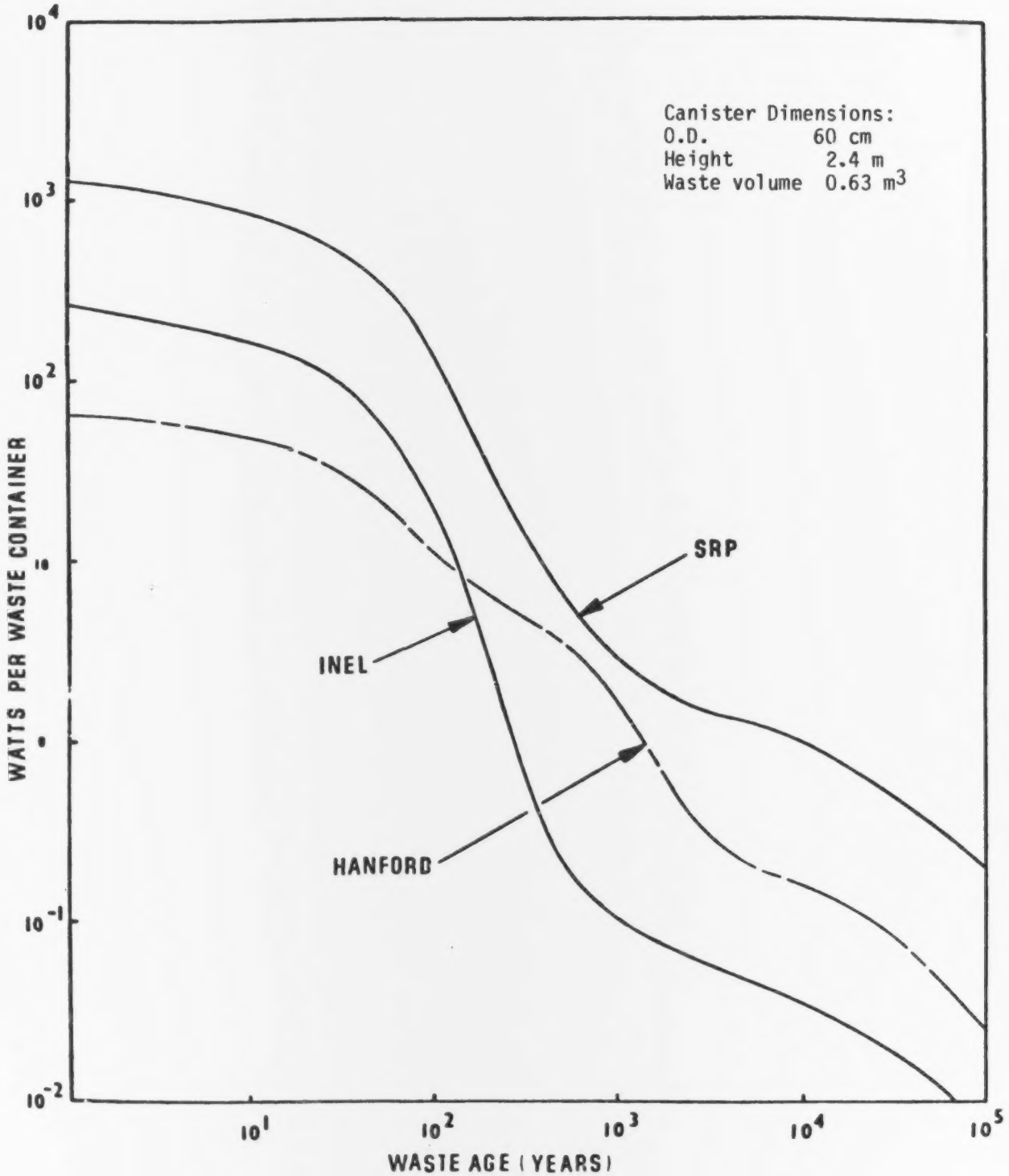

Figure 2.4. High level defense waste thermal histories.[20]

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Table 2.29

Chemical Composition of Retrieved Alumina and Zirconia Calcines at ICPP[22]

	Alumina Calcine			Zirconia Calcine		
	BIN 3 - Samples From:			BIN 7 - Samples From:		
	Top	Middle	Bottom	Top	Middle	Bottom
Major Constituents[a] (wt %)						
Al	35.53	35.93	39.75	7.53	7.63	7.62
Ca				25.49	29.14	27.86
F				20.10	20.30	21.10
Na	3.10	2.00	1.70			
NO_3	6.87	5.14	5.70			
Zr				16.87	17.52	17.74
Minor Constituents[b] (wt %)						
B	<0.2	<0.2	<0.2	0.60	0.68	0.65
Ca	0.92	0.92	3.54			
F	0.21	0.14	3.54			
Fe	<2.0	<2.0	<2.0	<2.0	<2.0	<2.0
Hg	1.6	1.8	1.9	0.05	0.02	0.04
K	0.5	0.5	0.4	0.10	0.06	0.08
Mn	<0.5	<0.5	<0.5	<0.05	<0.05	<0.05
Na				0.7	0.5	0.5
NO_3				1.03	0.71	2.88
Zr	<0.5	<0.5	<0.5			
Total rare earths	0.5	0.5	0.5	<0.5	<0.5	<0.5
Total Ru	<0.03	<0.04	<0.04	<0.05	<0.04	<0. 05

[a]Ranges of original WCF products are:
(1) WCF Hot Product - Al (40-48%), Na (0.9-1.5%), NO_3 (2.0%)
(2) WCF Run #4 Product - Al (7.0-9.9%), Ca (30.1-33.5%), Zr (15.7-17.4%), F (7.8-23.4%)

[b]Original WCF Run #4 Zirconia Calcine Product - Na (0.28-11.8%); information on other components not available.

Table 2.30

Radiochemical Analysis of Retrieved Alumina Calcine From Bin 3 at Idaho[22]

Sample From	Sr-90 g/g	Gamma Scan[a] Isotope	g/g	Uranium Mass g/g	Isotopic %		Transuranics Nuclide	g/g	
Top	38.0	Co-60	0.003	59.83	234	5.0	Pu-238	0.10	
		Cs-134	0.085		235	65.00	Pu-239	2.96	
		Cs-137	73.5		236	6.7	Pu-240	0.32	
		Eu-154	0.095		238	23.3	Pu-241	0.03	
							Pu-242	0.01	
							Np-237	92.1	5.2[b]
							Am-241	0.09	
Middle	40.5	Cs-134	0.004	53.16	234	1.1	Pu-238	0.10	
		Cs-137	73.5		235	69.4	Pu-239	2.98	
		Eu-154	0.133		236	3.7	Pu-240	0.32	
		Eu-155	0.0014		238	25.8	Pu-241	0.03	
							Pu-242	0.01	
							Np-237	190.6	5.9[b]
							Am-241	0.09	
Bottom	32.7	Cs-134	0.094	37.82	234	3.8	Pu-238	0.10	
		Cs-137	60.0		235	67.1	Pu-239	2.33	
		Eu-154	0.094		236	5.0	Pu-240	0.27	
					238	24.2	Pu-241	0.02	
							Pu-242	0.008	
							Np-237	182.4	4.0[b]
							Am-241	0.11	

[a]Fission products determined on 9/15/78.
[b]Recent data obtained from B. Staples, March 1980 (Ref. 23). The earlier data, reported in Ref. 22 are incorrect.

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Table 2.31

Radiochemical Analysis of Retrieved Zirconia Calcine From Bin 7 at Idaho[22]

Sample From	Sr-90 g/g	Gamma Scan[a] Isotope	g/g	Uranium Mass g/g	Isotopic %		Transuranics Nuclide	g/g	
Top	18.2	Co-60	0.002	51.11	234	3.4	Pu-238	1.11	
		Cs-134	0.027		235	68.9	Pu-239	3.89	
		Cs-137	21.7		236	6.0	Pu-240	0.98	
		Eu-154	0.137		238	21.7	Pu-241	0.26	
		Eu-155	0.0006				Pu-242	0.13	
							Np-237	95.7	1.3[b]
							Am-241	0.32	
Middle	17.5	Co-60	0.03	13.20	234	6.8	Pu-238	0.86	
		Cs-134	0.040		235	59.8	Pu-239	3.38	
		Cs-137	22.5		236	6.7	Pu-240	0.96	
		Eu-154	0.185		238	26.7	Pu-241	0.29	
		Eu-155	0.0009				Pu-242	0.17	
		Sb-125	0.019				Np-237	42.7	1.1[b]
							Am-241	0.24	
Bottom	15.0	Co-60	0.001	28.36	234	3.9	Pu-238	1.00	
		Cs-134	0.003		235	62.9	Pu-239	3.26	
		Cs-137	22.4		236	6.5	Pu-240	0.87	
		Eu-154	0.064		238	26.6	Pu-241	0.25	
		Eu-155	0.0010				Pu-242	0.13	
							Np-237	246.5	0.93[b]
							Am-241	0.18	

[a]Fission products determined on 9/15/78.
[b]Recent data obtained from B. Staples, March 1980 (Ref. 23). The earlier data, reported in Ref. 22 are incorrect.

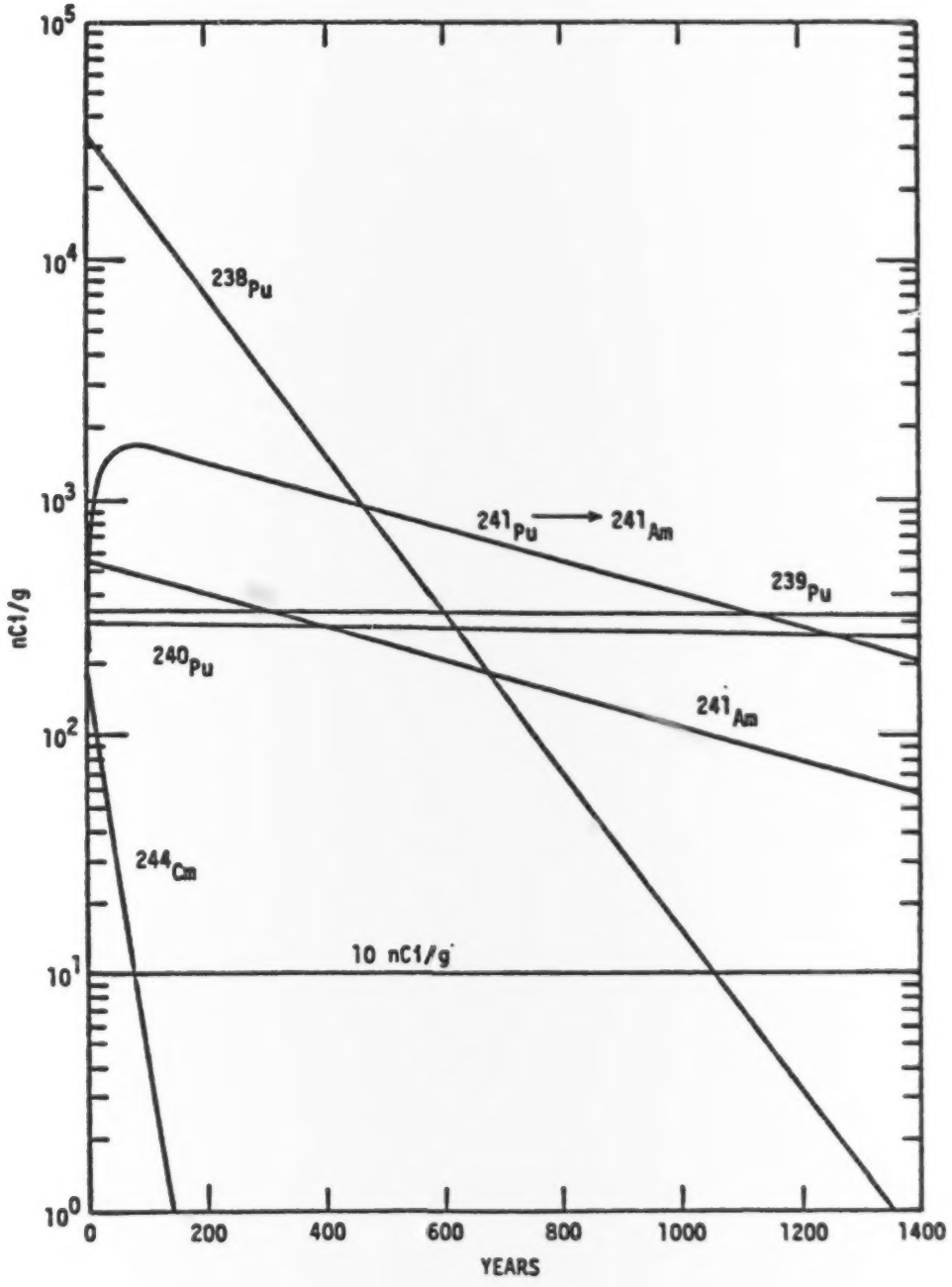

Figure 2.5. Significant actinides in ICPP calcine.[16]

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Table 2.32

Activity of Calcined Solids by Major Radionuclides
(As of October 1, 1979)

Radionuclide	Activity* 10^5 Ci		
	First Bin Set	Second Bin Set	Third Bin Set**
H-3		0.001	
Co-60		0.004	0.01
Sr-90	16.6	36.29	31.99
Ru-106			0.05
Sb-125		0.003	0.08
Cs-134	0.01	0.032	0.78
Cs-137	19.1	41.55	36.48
Ce-144		0.008	0.04
Pm-147	0.4	0.09	
Eu-154		0.023	0.17
TOTAL	36.11	78.00	69.60

* Radionuclide activities were estimated using liquid waste radionuclide analyses and volume of waste solution calcined decayed to October 1, 1979.

**Campaign is still in progress.

2.2.2 High Level Waste at Savannah River Plant

The Savannah River Plant (SRP) near Aiken, South Carolina, is at present the chief supplier of plutonium and tritium for the U.S. defense program. The Savannah River Plant has been processing its spent reactor fuel and targets since 1954. It has produced so far about 68 million gallons of high level waste, now concentrated to about 23 million gallons stored in 37 tanks.[26] The Savannah River Plant is currently producing waste at the rate of 1.5 million gallons per year.[26]

The liquid radioactive waste is produced at SRP primarily from nuclear fuel reprocessing operations in the F and H areas. The aqueous waste streams are generated from the PUREX process in the hot (high heat waste) and warm (low heat waste) canyons. These wastes are made alkaline and flow by gravity from the processing buildings to the tank farm through underground pipes. The processing of these high heat and low heat wastes in the tank farm is schematically described in Figure 2.6.[27] The high heat waste is aged for one to two years for the decay of the short lived fission products. After aging, the supernate is transferred to the evaporator. The concentrate from the evaporator goes to a cooled waste tank, where the suspended salts settle. The supernate from the salt storge tank goes back to the evaporator feed tank and the process continues until the liquid is converted to a crystalline salt cake. The low heat waste

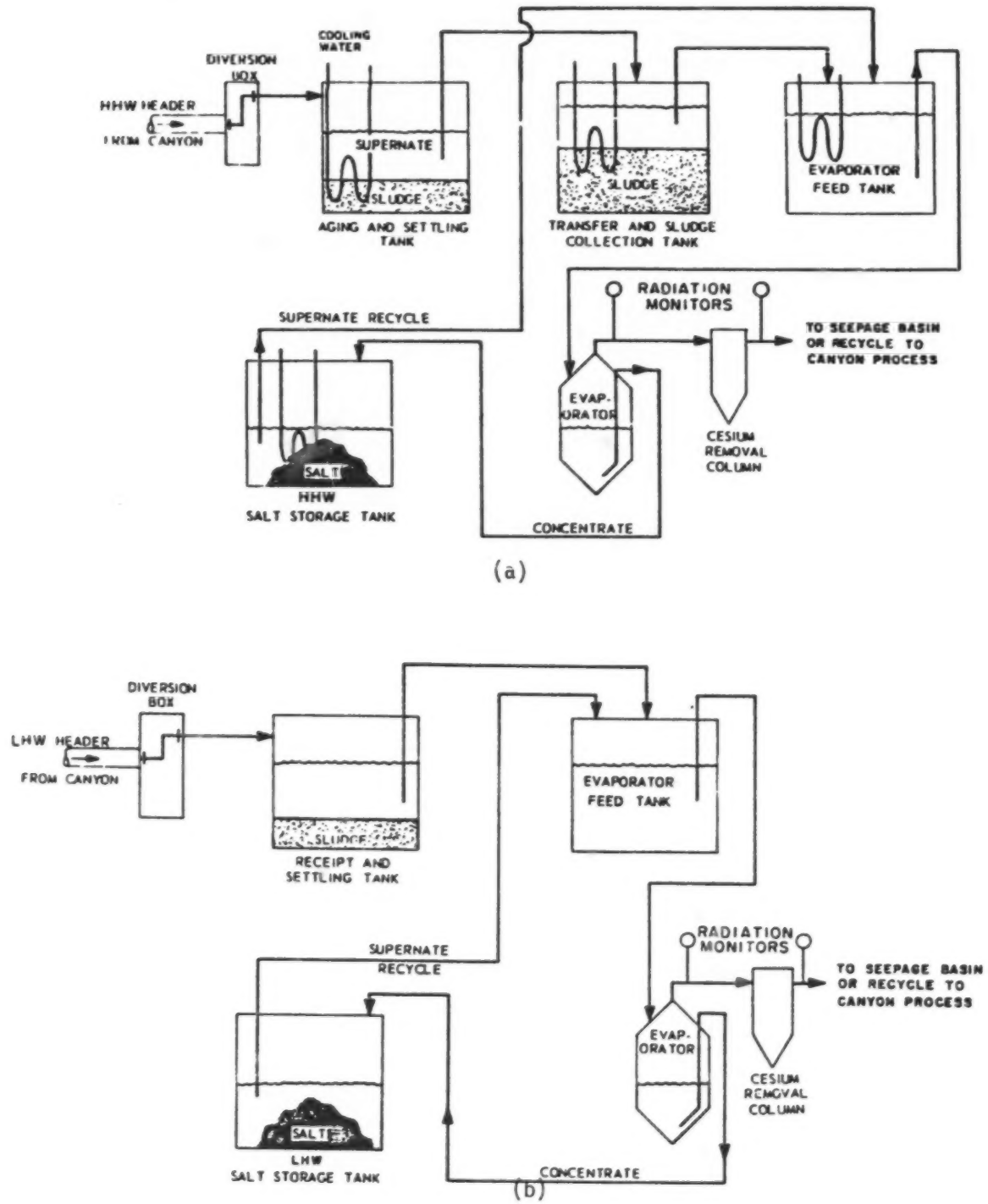

Figure 2.6. Schematic diagram of waste processing in-tank farm at SRP: (a) high heat waste, (b) low heat waste.[27]

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goes through a similar process except that it is not aged. Soluble salt cakes and separate sludges are thus produced at the Savannah River Plant and are stored in underground tanks. Low heat salts are kept in uncooled waste tanks. The chemical compositions of the low heat and high heat wastes are similar.

The Savannah River Plant has utilized 33 tanks for high level waste storage.[13,27] These tanks contain both the salt cake and the supernate and sludge. The tanks are located in the two farms (F and H). Each tank farm has one evaporator and one cesium removal column. As shown in Figure 2.6, these evaporators and columns concentrate waste and decontaminate condensate. Nineteen of these tanks are adjacent to one waste separation plant in the H area, and 14 are near the other separation plant in the F area. Four new tanks were scheduled for the period 1975-1978 for the F area. The underground tanks are built of carbon steel and reinforced concrete.[27] Reference 13 reports that there will be a total of 51 tanks operational by SRP by 1981 and no more tanks will be needed until 1987.

In addition to these tank farms, high level waste has been stored at the Receiving Basin for Offsite Fuel (RBOF). The fuel came from different types of reactors, most of which have since been shut down. This basin is usually used for short term storage of spent fuel. However, some fuel has been stored here for years.[13] The amount of percentage of SRP in high level waste at the RBOF is not known to us at present.

There has been no intentional release of high level waste to soils. However, there was one unintentional release of a few tens of gallons of high level waste from a tank into the soil in the 1959 to 1960 period.[13]

Existing and projected through 1984 volumes of high level waste at the Savannah River site are shown in Table 2.33. Approximate volumes of liquid, salt and sludge in each of the 33 high level waste tanks at the Savannah River plant are shown in Figures 2.7 and 2.8.[26] The NRC document[13] quotes the total volume of stored high level waste in the tank farms to be 8.33×10^4 m^3 or 22×10^6 gallons, or 2942×10^3 ft^3, which agrees with Table 2.33. The Savannah River document (Ref. 28) reports that the total volume will be kept the same through 1980, although the curies of radioactivity will increase. Latest data obtained from Savannah River Laboratory[26,29] indicate that the present SRP volume of high level waste is 23 million gallons. Details of sludge volumes in the F and H area tanks are also obtained and are given in Table 2.34. Figure 2.9 shows the inventory of the high level waste at SRP from 1961-1974.[11]

The total amount of high level radioactivity present in the Savannah River tanks is estimated to be $7.5. \times 10^8$ curies by the NUREG document.[13] Reference 25 reports the total inventory to be 5.7×10^8 curies in 1978 and 6.8×10^8 curies in 1990. Approximately, then, about 30 curies/gallon (230 curies/cubic ft) of radioactivity is stored in the tanks at the SRP. The amount of plutonium in the Savannah River waste is estimated to be 225 kg.[30]

The liquid high level waste composition varies from tank to tank. The average chemical composition of the fresh and the reconstituted high level

Table 2.33

Existing and Projected Volumes of High Level Wastes at Savannah River Plant

Waste Composition	Million gallons					
	Ref. 26*			Ref. 6	Ref. 6	Ref. 25
	F-Area	H-Area	Total	as of 10/1/77	as of 1/1/85	
Liquid	5.3	7.5	12.8	12.72	4.19	5.55
Salt	4.0	2.5	6.5	6.73	12.42	13.19
Sludge	1.2	1.6	2.8	2.24	3.07	3.37
TOTAL	10.5	11.6	22.1	21.69	19.68	22.11

*Only 33 tanks were included in this estimate as shown in Figures 2.7 and 2.8.

Table 2.34

General Composition of SRP Sludge[26]

Sludge Type	Source	Estimated Quantity, 10^3 Gallons		
		F	H	Total
High Fe	Purex Haw HM LAW	388	390	778
High Al	HM HAW	---	975	975
High Al	Reconstituted Salt Cake[a]	353	235	588
Mixed Al, Fe	Purex LAW HAW/LAW Blend	930	164	1094
Zeolite	CRC	13	32	45
Other	RBOF/RRF	---	32	32
TOTAL		1684	1828	3512

[a]Now included in salt inventory; estimated based on two salt samples.

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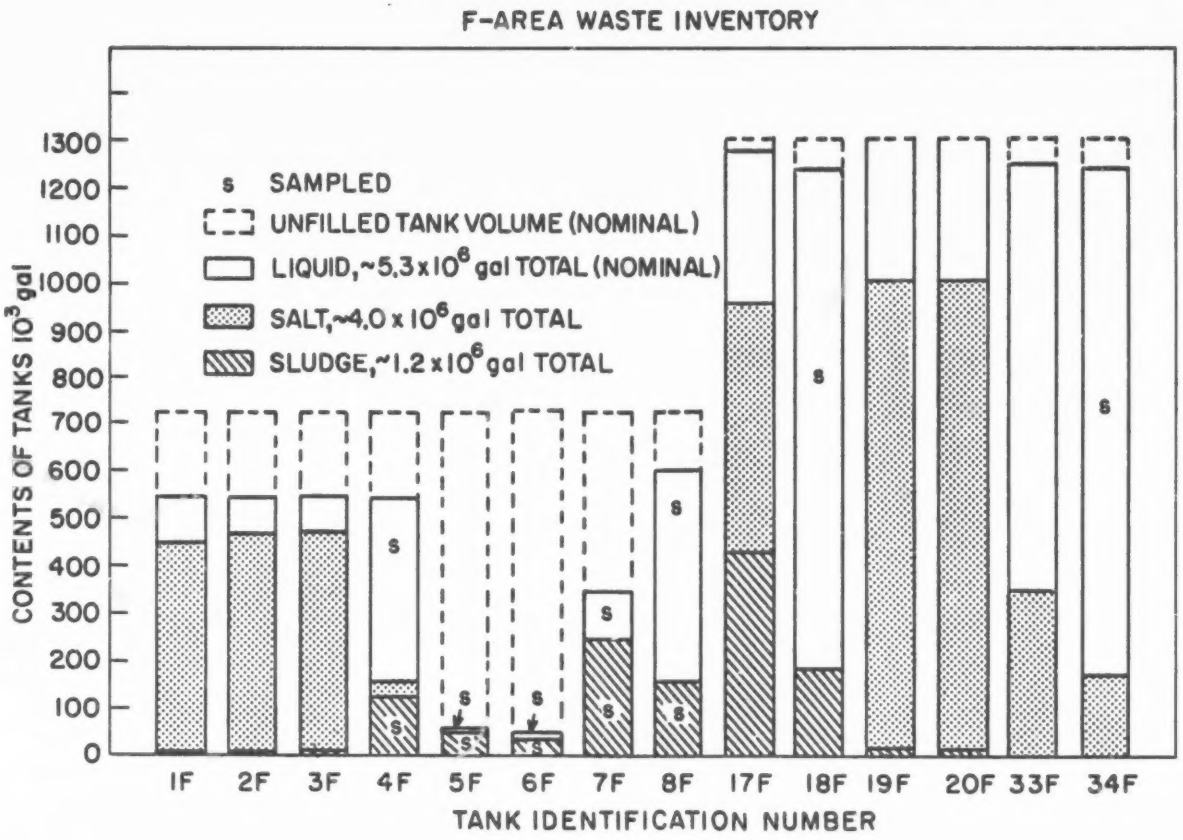

Figure 2.7. Savannah River Plant high level waste volumes - F Area.[26]

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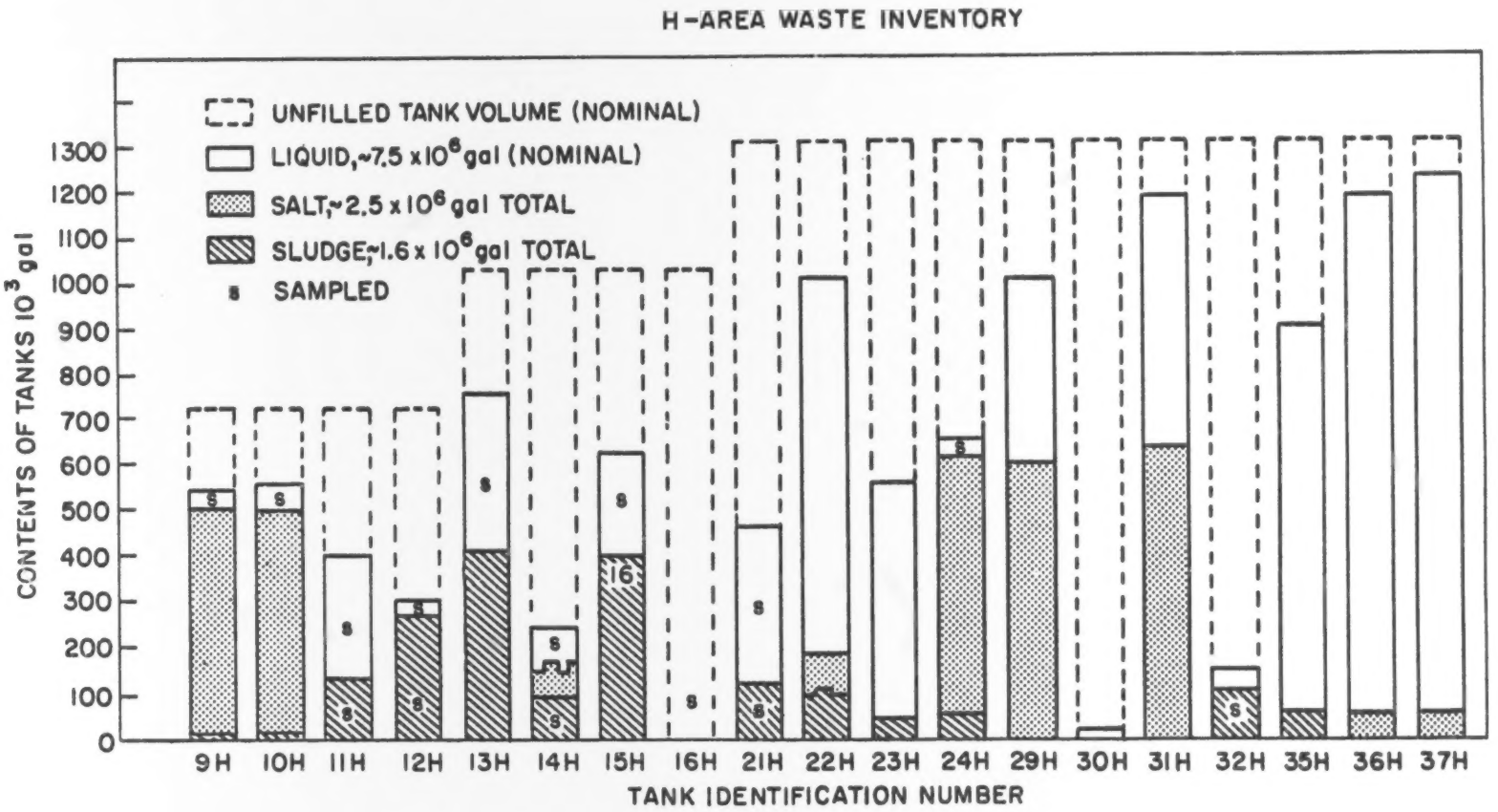

Figure 2.8. Savannah River Plant high level waste volumes - H Area.[26]

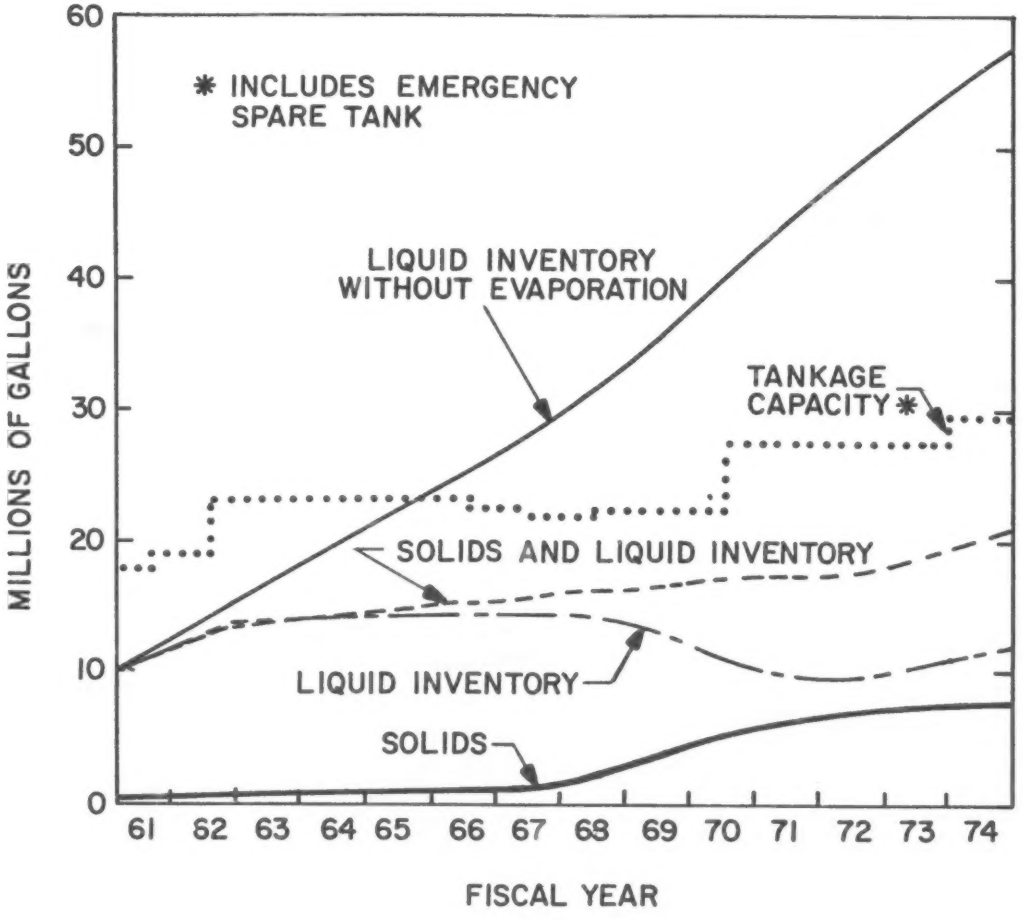

Figure 2.9. Savannah River high level waste inventory from 1961-1974 (from GAO Document No. RED-75-309).

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liquid waste in the tanks at SRP are shown in Table 2.35.[31] The chemical composition of the waste changes with time because of radiolytic decomposition and chemical reactions. Thus, $NaNO_3$ decreases with an equivalent increase in $NaNO_2$ by radiolytic reduction. NaOH reacts with CO_2 from air and from Na_2CO_3; NaOH is added periodically to maintain its concentration. This is reflected in Table 2.35. The variations in concentration of the constituents in the SRP supernates are shown[32] in Table 2.36. Some further details of the supernate major constituents are shown separately in Table 2.37 for the F-area and H-area tank farms.

The sludge in a tank contains mostly oxides and hydroxides of manganese, iron, and aluminum. It has essentially all of the fission products of the irradiated fuel except cesium. The sludge also contains a significant quantity of mercury. The average chemical composition of the SRP sludges and their ranges are shown in Table 2.38. Details of the sludge composition in SRP Tank 15 are shown in Table 2.39 for three different times.[26] A comparison of the second and third columns in this table reveal a rather significant time-dependence of these compositions. The accuracies of these chemical compositions are not known. Compositions of SRP sludge calcines are shown in Table 2.40. This is obtained from Lawrence Livermore Laboratory work.[33] This table clearly indicates the tank-to-tank compositional variability of sludge at the SRP. The high iron and high aluminum compositions in this table are obtained from specific tanks that represent these extremes.

Table 2.35

Average Chemical Composition of SRP High Level Waste[31]

Constituents	Concentration[25] Fresh (M)	Reconstituted[b] (M)	Wt(%)[c]
$NaNO_3$	3.3	2.2	31.6
$NaNO_2$	<0.2	1.1	12.8
$NaAl(OH)_4$	0.5	0.5	10.0
NaOH	1.0	0.75	5.1
Na_2CO_3	0.1	0.3	5.4
Na_2SO_4	0.3	0.3	7.2
$Fe(OH)_3$	0.07	0.07	1.3
MnO_2	0.02	0.02	0.3
$Hg(OH)_2$	0.002	0.002	0.1
Other Solids[a]	0.13	0.13	1.3
Total			75.1

[a]Assumes an average molecular weight of 60.
[b]Reference year not known, but probably 1985.
[c]Calculated from molar concentrations with the assumption of 25% of H_2O.

Table 2.36

Ranges of Chemical Composition of SRP Supernates[31]

Constituent	Concentration (M)
$NaNO_3$, $NaNO_2$	3 - 6
Na_2CO_3	0.1 - 0.3
Na_2SO_4	0.02 - 0.2
NaOH	1 - 6
$NaAl(OH)_4$	0.4 - 1.0

Table 2.37

Concentration Range of Major Constituents in SRP Supernate[26]

Constituent	Molarity Ranges	
	F-Area	H-Area
Na^+	4.0-12.5	5.7-12.5
NO_3^-	1.6-2.5	1.9-6.4
NO_2^-	0.5-3.1	0.2-3.2
$Al(OH)_4^-$	0.4-0.8	0.4-1.6
OH^-	1.1-6.3	0.8-3.8

Table 2.38

Chemical Composition of SRP Sludges[32]
(wt %)

Component	Average	Range
Fe [$Fe(OH)_3$]	22.0	3.1 - 32.8
Al	9.9	1.5 - 33.5
U	6.7	0 - 15.4
Mn (MnO_2)	5.4	1.7 - 10.8
Na	3.3	1.0 - 6.1
Ni	2.8	0.3 - 6.3
Ca	1.6	0.2 - 2.9
Hg	1.2	0.1 - 2.8
$NO_3^- + NO_2^-$	---	0.1 - 1.3
SO_4^{-2}	0.5	---

Table 2.39

SRP Tank 15 Sludge Analysis[26]

Constituent	Wt. Percent		
	10/74	8/75	4/79
Al	33.5	19.7	20.5
Fe	3.0	5.6	8.0
Hg	0.9	2.6	9.4
Mn	2.2	2.6	0.8
Na	1.2	2.6	1.6
Ca	0.2	0.6	0.4
Cr	<0.1		0.3
Th	0.2		2.7
U	0.9	4.0	0.3

Table 2.40

Major Components in SRP Waste Calcines[33]

Component	High Fe	Composite	High Al
Fe_2O_3	53.17	36.13	5.32
Al_2O_3	4.89	28.26	76.05
MnO_2	3.56	9.94	4.37
U_3O_8	12.34	3.26	1.28
CaO	3.62	2.69	0.35
NiO	9.08	4.47	0.78
SiO_2	0.40	0.85	0.56
Na_2O	4.52	5.08	1.96
Na_2SO_4	<0.50	0.93	<0.50
Ion-Siv*	8.82	8.93	9.33

*Ca-K-Na zeolite ion exchange medium.

The salt cake consists primarily of $NaNO_3$, NaOH, Na_2CO_3 and $NaAl(OH)_4$. This is obtained by evaporating the supernate and then cooling to crystallize. The chemical composition of salt cake in Tanks 19 and 20 are given in Table 2.41.[26] The radionuclide concentration in the salt is approximately three times that of the supernate.

Table 2.41

SRP Tanks 19 and 20 Salt Cake Analyses[26]

Component	Wt. Percent	
	Tank 19	Tank 20
$NaNO_3$	50.1	77.2
$NaNO_2$	0.9	0.9
$NaAl(OH)_4$	14.5	0.6
NaOH	4.1	2.5
Na_2CO_3	3.4	1.9
Na_2SO_4	12.0	6.4
NaF	0.1	0.2
NaCl	interf.	interf.
$Na[HgO(OH)]$	n.m.	n.m.
$Na_2C_2O_4$	0.1	0.1
Insoluble	4.0	3.1

n.m. = not measured.

The radionuclide characteristics of the Savannah River waste are calculated by the Savannah River staff with the help of a code called TAFI,[34] which is a depletion calculation code. The concentrations of radionuclides in the fresh waste are shown in Table 2.42.[35] Figure 2.10 shows the composition and decay of the radionuclides in the waste.[31] The range of concentration of Cs-137 in the SRP supernate varies from 0.1 to 20 Ci/gal, respectively.[27] The average radionuclide composition of SRP waste in the sludge and in the supernate are shown in Table 2.43 and 2.44, respectively.[27] These are given at 1, 5, and 10 years after irradiation. The most active isotopes after 10 years are Sr-90, Cs-137, Pm-147, Sm-151, Cs-134, of which Sr-90 contributes 24 Ci/gal in the sludge, and Cs-137 contributes 4 Ci/gal in sludge and supernate together. This picture changes after a few hundred years, at which stage Sm-151, Pu-238, Tc-99, Pu-239, Am-241, Zr-93, Pu-240, Cs-135, etc., start to dominate. The relative composition and activity in curies/gallon of different isotopes in terms of reconstituted waste is shown in Figures 2.11 and 2.12 from 0 to 1400 years and from 0 to 10^6 years, respectively.[31] The reconstituted waste (supernate sludge and dissolved salt all together in one solution) at SRP will produce 60 million gallons of waste and will have the chemical composition as shown in Table 2.35. Table 2.45 shows their radionuclide content[31] for 1985. This is the waste to be considered for solidification programs.[31]

Sutherland and Bennett[20] have calculated the worst activity and heat generation rates to be expected if the Savannah River high level waste were converted to a glass matrix form.* The isotopic composition of their final

*The data for this "worst" or "hottest" waste were obtained by Sutherland from the Savannah River Laboartory.[35]

Table 2.42

Average Radionuclide Composition of Fresh[a] SRP High Level Waste[31]

Radionuclide	Activity, Ci/gal	Radionuclide	Activity, Ci/gal
Ce-144-Pr-144	68	Am-241	1×10^{-3}
Zr-95	60	Tc-99	5×10^{-4}
Y-91	47	Pu-239	3×10^{-4}
Sr-89	36	Eu-154	1×10^{-4}
Nb-95	15	Zr-93	1×10^{-4}
Ce-141	12	Pu-240	6×10^{-5}
Pm-147	12	Cs-135	4×10^{-5}
Ru-103	10	Sn-126-Sb-126	1×10^{-5}
Ru-106-Rh-106	4	Se-79	1×10^{-5}
Sr-90	3	U-233	2×10^{-6}
Cs-137	3	I-129	1×10^{-6}
Te-129	2	U-238	6×10^{-7}
Te-127	2	Pd-107	5×10^{-7}
Cs-134[b]	1	Np-237	4×10^{-7}
Sm-151	8×10^{-2}	Eu-152	2×10^{-7}
Pu-238	1×10^{-2}	Pu-242	6×10^{-8}
Pu-241	2×10^{-3}	Tb-158	6×10^{-8}
Cm-244	1×10^{-3}	U-235	3×10^{-8}

[a]After reprocessing fuel that has been cooled six months after discharge from reactor.
[b]This has been reported in various sources as Ce-134; however, the data indicates that this should be Cs-134.

waste form is shown in Table 2.46. These numbers were generated by Sutherland and Bennett with the help of the SANDIA-ORIGEN code[21] from existing DOE data.[31,35] The numbers in Table 2.46 appear to be much higher compared to the activity of the real waste as shown in Table 2.44. The time dependence of the total activity and the heating rates in the final waste form, as obtained by Sutherland and Bennett are shown in Table 2.47. The curie content and the heating rates for this waste are also graphically depicted in Figures 2.3 and 2.4 for a specific canister design. The activity and the heating rates in the SRP waste are higher compared to the Hanford wastes because Sr and Cs were taken out from the Hanford wastes. Thus the SRP wastes are the most radioactive wastes among the three types of defense wastes.

It is strongly recommended that an experimental validation of the SRP inventory be made since the volume is large and operation of the facility will continue to generate additional high level wastes.

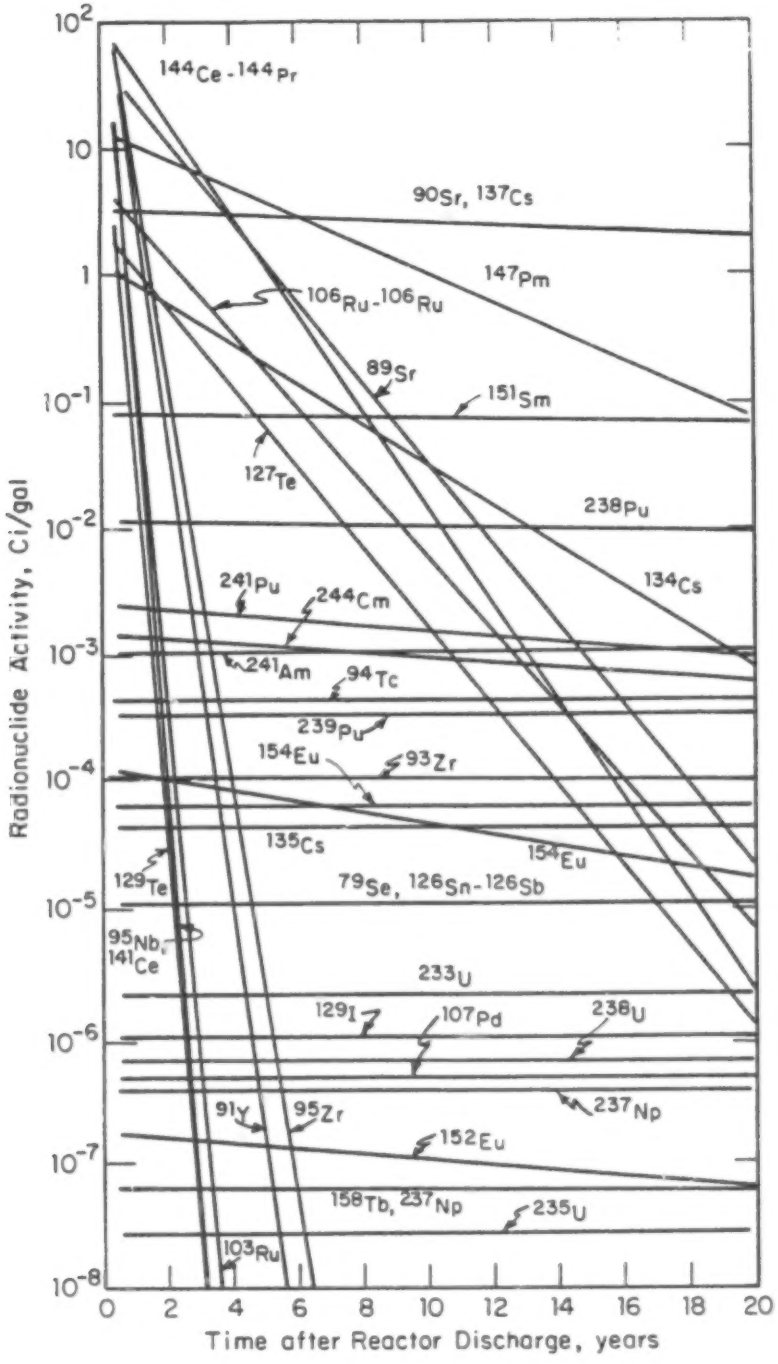

Figure 2.10. Radionuclide composition of SRP fresh waste.

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Table 2.43

Average Radionuclide Composition of SRP High-Level Supernate[31]

	Radionuclide Activity, Ci/gal		
Time After Irradiation, yrs	1	5	10
Ce-144-Pr-144	2.6	7.4×10^{-2}	8.7×10^{-4}
Zr-95	2.7	5.0×10^{-7}	a
Y-91	1.7×10^{-1}	5.7×10^{-5}	a
Sr-89	1.0×10^{-1}	a	a
Nb-98	1.7×10^{-1}	a	a
Ce-141	1.7×10^{-1}	a	a
Pm-147	6.1×10^{-1}	2.1×10^{-1}	5.7×10^{-2}
Ru-103	1.4×10^{-1}	a	a
Ru-106-Rh-106	6.7×10^{-1}	4.3×10^{-2}	1.4×10^{-3}
Sr-90	6.8×10^{-2}	6.2×10^{-2}	5.5×10^{-2}
Cs-137	3.3	3.1	2.7
Te-129	5.5×10^{-3}	a	a
Te-127	3.8×10^{-2}	3.4×10^{-6}	a
Cs-134[b]	5.1×10^{-2}	1.3×10^{-2}	2.4×10^{-3}
Sm-151	4.4×10^{-3}	4.3×10^{-3}	4.1×10^{-3}
Pu-238	3.8×10^{-5}	3.7×10^{-5}	3.5×10^{-3}
Pu-241	8.1×10^{-6}	6.7×10^{-6}	5.4×10^{-6}
Cm-244	4.5×10^{-6}	3.8×10^{-6}	3.2×10^{-6}
Am-241	3.6×10^{-6}	3.6×10^{-6}	3.6×10^{-6}
Tc-99	2.5×10^{-5}	2.5×10^{-5}	2.5×10^{-5}
Pu-239	1.1×10^{-6}	1.1×10^{-6}	1.1×10^{-6}
Eu-154	6.7×10^{-6}	4.8×10^{-6}	3.2×10^{-6}
Zr-93	2.4×10^{-5}	2.4×10^{-5}	2.4×10^{-5}
Pu-240	2.1×10^{-7}	2.1×10^{-7}	2.1×10^{-7}
Cs-135	4.6×10^{-5}	4.6×10^{-5}	4.6×10^{-5}
Sn-126-Sb-126	6.1×10^{-7}	6.1×10^{-7}	6.1×10^{-7}
Se-79	6.0×10^{-7}	6.0×10^{-7}	6.0×10^{-7}
U-233	7.1×10^{-9}	7.1×10^{-9}	7.1×10^{-9}
I-129	5.5×10^{-8}	5.5×10^{-8}	5.5×10^{-8}
U-238	2.1×10^{-9}	2.1×10^{-9}	2.1×10^{-9}
Pd-107	2.6×10^{-8}	2.6×10^{-8}	2.6×10^{-8}
Np-237	1.3×10^{-9}	1.3×10^{-9}	1.3×10^{-9}
Eu-152	1.0×10^{-8}	7.8×10^{-9}	6.0×10^{-9}
Pu-242	a	a	a
Tb-158	a	a	a
U-235	a	a	a

[a]Value $< 1 \times 10^{-9}$

[b]This has been reported in various sources as Ce-134; however, the data indicates that this should be Cs-134.

Table 2.44

Average Radionuclide Composition of SRP High-Level Sludge[31]

Time After Irradiation, yrs	Radionuclide Activity, Ci/gal		
	1	5	10
Ce-144-Pr-144	4.5×10^{2}	1.3×10^{1}	1.5×10^{-1}
Zr-95	9.6×10^{1}	1.8×10^{-5}	a
Y-91	7.6×10^{1}	2.5×10^{-6}	a
Sr-89	4.4×10^{1}	a	a
Nb-98	6.0×10^{0}	a	a
Ce-141	3.0×10^{0}	a	a
Pm-147	1.0×10^{2}	3.6×10^{1}	9.7×10^{0}
Ru-103	5.2×10^{0}	a	a
Ru-106-Rh-106	2.4×10^{1}	1.6×10^{0}	5×10^{-2}
Sr-90	3.0×10^{1}	2.8×10^{1}	2.4×10^{1}
Cs-137	1.6×10^{0}	1.5×10^{0}	1.3×10^{0}
Te-129	9.4×10^{-1}	a	a
Te-127	6.4×10^{0}	5.9×10^{-4}	a
Cs-134[b]	8.7×10^{0}	2.3×10^{0}	4.2×10^{-1}
Sm-151	7.5×10^{-1}	7.3×10^{-1}	7.0×10^{-1}
Pu-238	1.1×10^{-1}	1.1×10^{-1}	1.1×10^{-1}
Pu-241	2.4×10^{-2}	2.0×10^{-2}	1.6×10^{-2}
Cm-244	1.3×10^{-2}	1.1×10^{-2}	9.5×10^{-3}
Am-241	1.1×10^{-2}	1.1×10^{-2}	1.1×10^{-2}
Tc-99	4.3×10^{-3}	4.3×10^{-3}	4.3×10^{-3}
Pu-239	3.5×10^{-3}	3.5×10^{-3}	3.5×10^{-3}
Eu-154	1.1×10^{-3}	8.3×10^{-4}	5.5×10^{-4}
Zr-93	8.6×10^{-4}	8.6×10^{-4}	8.6×10^{-4}
Pu-240	6.4×10^{-4}	6.4×10^{-4}	6.4×10^{-4}
Cs-135	2.2×10^{-5}	2.2×10^{-5}	2.2×10^{-5}
Sn-126-Sb-126	1.1×10^{-4}	1.1×10^{-4}	1.1×10^{-4}
Se-79	1.0×10^{-4}	1.0×10^{-4}	1.0×10^{-4}
U-233	2.1×10^{-5}	2.1×10^{-5}	2.1×10^{-5}
I-129	9.4×10^{-6}	9.4×10^{-6}	9.4×10^{-6}
U-238	6.4×10^{-6}	6.4×10^{-6}	6.4×10^{-6}
Pd-107	4.4×10^{-6}	4.4×10^{-6}	4.4×10^{-6}
Np-237	3.9×10^{-6}	3.9×10^{-6}	3.9×10^{-6}
Eu-152	1.7×10^{-6}	1.3×10^{-6}	1.0×10^{-6}
Pu-242	6.2×10^{-7}	6.2×10^{-7}	6.2×10^{-7}
Tb-158	6.0×10^{-7}	6.0×10^{-7}	6.0×10^{-7}
U-235	2.7×10^{-7}	2.7×10^{-7}	2.7×10^{-7}

[a] Values $< 1 \times 10^{-7}$

[b] This has been reported in various sources as Ce-134; however, the data indicates that this should be Cs-134.

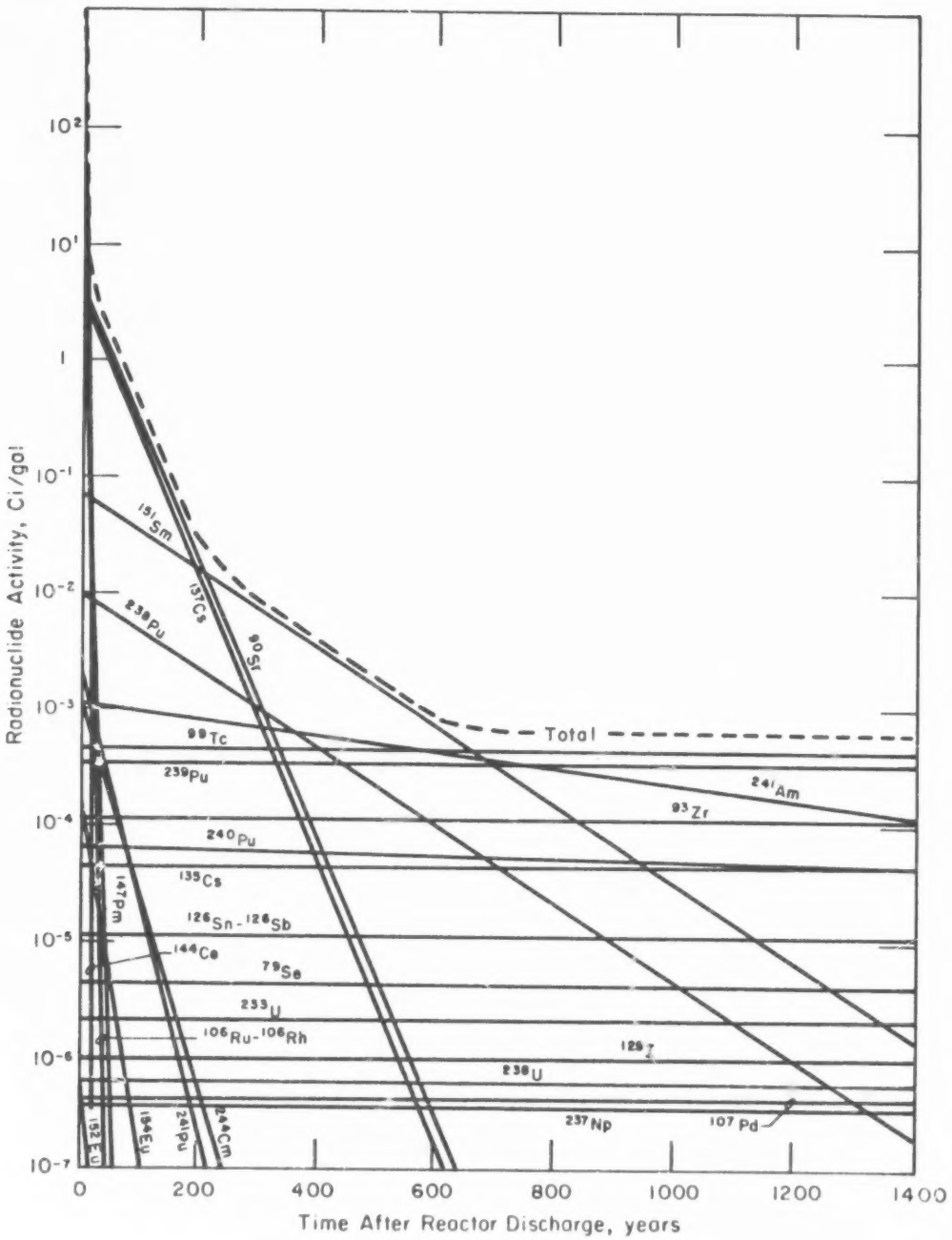

Figure 2.11. Radionuclide composition of SRP reconstituted waste: 0 to 1400 years after irradiation.[31]

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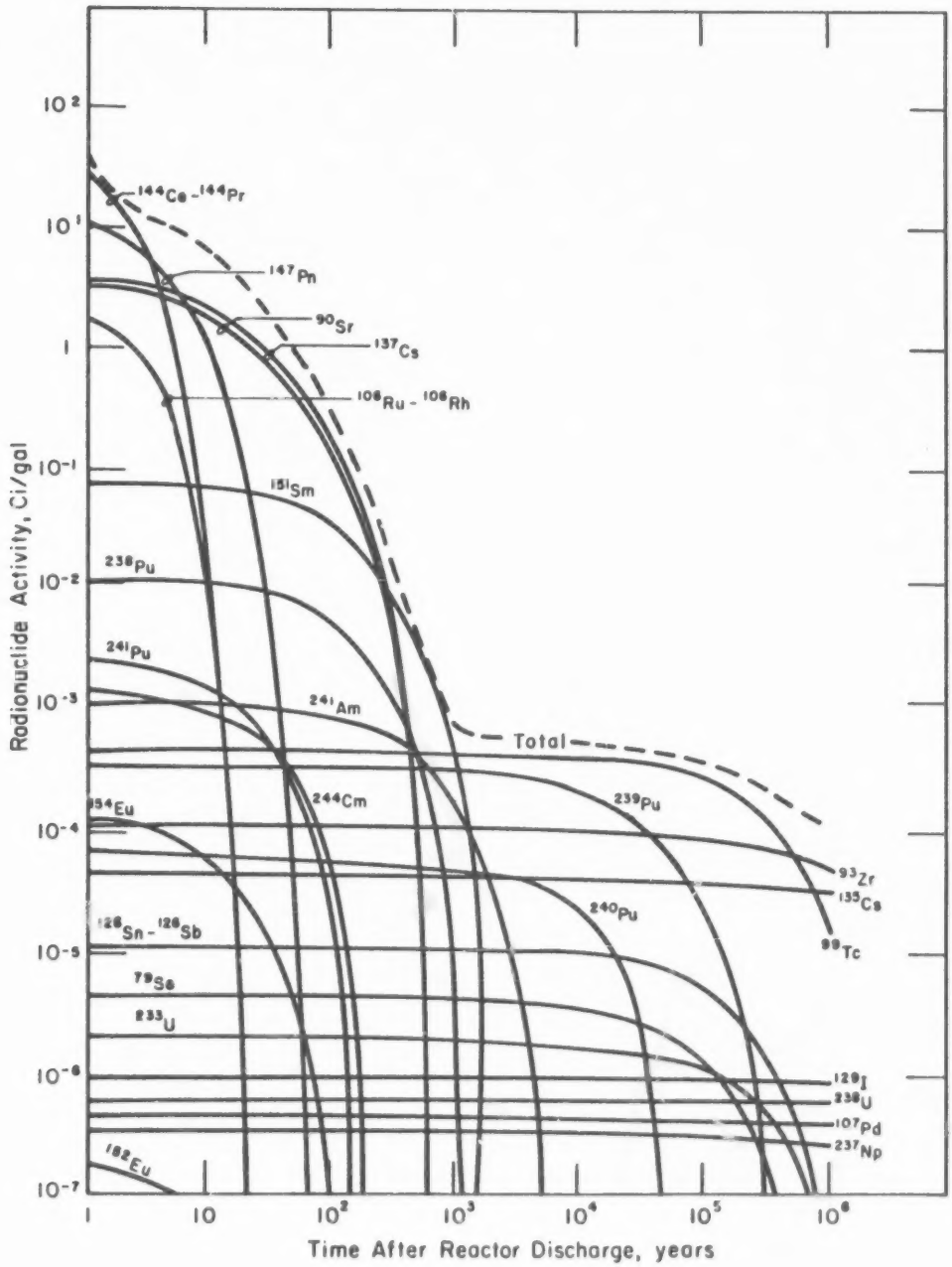

Figure 2.12. Radionuclide composition of SRP reconstituted waste: 1 to 10^6 years after irradiation.[31]

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Table 2.45

Radionuclide Content of Reconstituted SRP High Level Waste (1985)[31]

Radionuclide	Activity, Ci/gal	Total Activity, Ci
Sr-90	2.1	1.3×10^8
Cs-137	2.2	1.3×10^8
Pm-147	0.77	4.6×10^7
Ce-144-Pr-144	0.19	1.1×10^7
Sm-151	0.07	4.2×10^6
Ru-106-Rh-106	0.03	1.8×10^6
Pu-238	0.01	6.0×10^5
Am-241	0.001	6.0×10^4
Cm-244	0.001	6.0×10^4
Pu-239	0.0004	2.4×10^4

Table 2.46

Radionuclide Content of Savannah River Defense High Level Waste in Glass Form[20]

Isotope	Curies/m^3*	Isotope	Curies/m^3
Co-60	3.96 + 2**	Pm-147	4.63 + 4
Se-79	3.39 - 1	Sm-151	1.35 + 2
Sr-90	7.51 + 4	Eu-152	5.98 + 0
Y-90	7.83 + 4	Eu-154	3.26 + 3
Y-91	2.72 - 4	Eu-155	3.72 + 2
Zr-93	2.40 + 0	Tl-208	1.82 + 0
Zr-95	3.04 - 3	U-232	1.49 + 0
Nb-95	6.62 - 3	U-233	6.39 - 7
Ru-106	2.54 + 3	U-234	4.23 - 2
Rh-106	2.56 + 3	U-235	2.14 - 5
Pd-107	2.27 - 2	U-236	2.72 - 4
Ag-110	3.99 + 1	U-238	2.14 - 4
Sn-117m	1.43 - 1	Np-237	1.31 - 3
Sn-121m	5.19 + 1	Pu-236	1.63 + 1
Sn-123	2.80 - 1	Pu-238	2.78 + 3
Sn-126	3.04 - 2	Pu-239	3.10 + 1
Sb-125	1.43 + 3	Pu-240	2.43 + 1
Sb-126m	1.38 - 1	Pu-241	4.43 + 3
Sb-126	1.38 - 1	Pu-242	3.45 - 2
Te-125m	3.31 + 2	Am-241	4.84 + 1
Te-127m	5.54 - 2	Am-242	3.56 - 2

Table 2.46 Continued

Isotope	Curies/m^3*	Isotope	Curies/m^3
Te-127	5.63 - 2	Am-242m	3.56 - 2
Cs-134	2.21 + 4	Am-243	1.12 - 1
Cs-135	1.92 - 1	Cm-242	7.80 - 2
Cs-137	8.49 + 4	Cm-243	2.89 - 2
Ba-137m	8.18 + 4	Cm-244	3.15 + 0
Ce-142	2.54 - 5	Cm-245	1.44 - 4
Ce-144	1.74 + 4	Cm-246	1.34 - 5
Pr-144	1.77 + 4		

* Ci/m^3 of the final product.
**3.96 + 2 = 3.96 x 10^2.

Table 2.47

Savannah River Defense High Level Waste Curie and Heat Contents[20]

Waste Age (yr)	Curies/m^3	Watts/m^3	Waste Age (yr)	Curies/m^3	Watts/m^3
1	3.88 + 5	1.30 + 3	600	1.64 + 2	5.00 + 0
2	3.57 + 5	1.18 + 3	700	1.37 + 2	4.14 + 0
3	3.36 + 5	1.09 + 3	800	1.20 + 2	3.60 + 0
4	3.19 + 5	1.03 + 3	900	1.09 + 2	3.22 + 0
5	3.06 + 5	9.88 + 2	1000	1.00 + 2	2.95 + 0
6	2.95 + 5	9.50 + 2	2000	6.56 + 1	1.80 + 0
7	2.85 + 5	9.17 + 2	3000	5.68 + 1	1.52 + 0
8	2.76 + 5	8.89 + 2	4000	5.31 + 1	1.40 + 0
9	2.67 + 5	8.63 + 2	5000	5.06 + 1	1.32 + 0
10	2.60 + 5	8.41 + 2	6000	4.85 + 1	1.25 + 0
20	2.01 + 5	6.65 + 2	7000	4.65 + 1	1.19 - 0
30	1.58 + 5	5.35 + 2	8000	4.48 + 1	1.14 - 0
40	1.25 + 5	4.34 + 2	9000	4.31 + 1	1.09 - 0
50	9.88 + 4	3.54 + 2	10000	4.16 + 1	1.04 - 0
60	7.82 + 4	2.90 + 2	20000	3.10 + 1	7.06 - 1
70	6.19 + 4	2.39 + 2	30000	2.53 + 1	5.25 - 1
80	4.91 + 4	1.98 + 2	40000	2.19 + 1	4.15 - 1
90	3.90 + 4	1.65 + 2	50000	1.96 + 1	3.42 - 1
100	3.10 + 4	1.39 + 2	60000	1.80 + 1	2.91 - 1
200	3.63 + 3	3.58 + 1	70000	1.69 + 1	2.56 - 1
300	7.51 + 2	1.62 + 1	80000	1.62 + 1	2.30 - 1
400	3.28 + 2	9.52 + 0	90000	1.56 + 1	2.12 - 1
500	2.14 + 2	6.53 + 0	100000	1.52 + 1	1.99 - 1

2.2.3. High Level Waste at Hanford Reservation Operations

A large amount of DOE high level radioactive waste is stored at the Hanford Reservation in Richland, Washington. It has about 70% of the AEC generated radioactive waste volume.[11,18] Elaborate details of the waste management operations at Hanford are available in Reference 36. Information available on the inventory of the high level radioactive waste at Hanford has been assembled from several sources and is described below.

High level radioactive waste has been accumulating at Hanford since 1944 when the first reactor fuel was processed for plutonium recovery. The waste containing most of the fission products and actinides was originally stored in underground single shell storage tanks (steel-lined concrete tanks). Because of leakage of radioactive liquid waste from some of the tanks, the waste management policy changed to reduce the volume of liquid waste by evaporation of water to form salt and residual liquor. Since 1968, the high heat emitting isotopes, Sr-90 and Cs-137 are being removed from the waste by a fractionization process and converted to solid strontium fluoride (SrF_2) and cesium chloride (CsCl) compounds. These are placed in double walled capsules for storage in water basins. The liquid waste remaining after removal of Sr-90 and Cs-137 is made alkaline and returned to the underground tanks for eventual evaporation to a solid salt cake. The current Hanford radioactive waste management program is described in Figure 2.13.[37]

The radioactive waste at Hanford was generated mostly by a total of eight graphite moderated, once-through water cooled reactors and by the dual purpose N-reactor. Only the N-reactor is currently in operation.[36] The status of these reactors is given in Table 2.48.[36] The reactor facilities are located along the Columbia River in the 100 Areas.

Table 2.48

Hanford Reactor Status Summary[36]

Area	Size-Acres	Reactors	Construction Start	Operation Start	Operation Shutdown
100-B/C	650	105-B	Aug 1943	Sept 1944	Feb 1968
		105-C	June 1951	Nov 1952	April 1969
100-K	135	105-KW	Nov 1952	Jan 1955	Feb 1970
		105-KE	Jan 1953	April 1955	Jan 1971
100-N	90	105-N	May 1959	Dec 1963	Operating
100-D/DR	960	105-D	Nov 1943	Dec 1944	June 1967
		105-DR	Dec 1947	Oct 1950	Dec 1964
100-H	320	105-H	Mar 1948	Oct 1949	April 1965
100-F	540	105-F	Dec 1943	Feb 1945	June 1965

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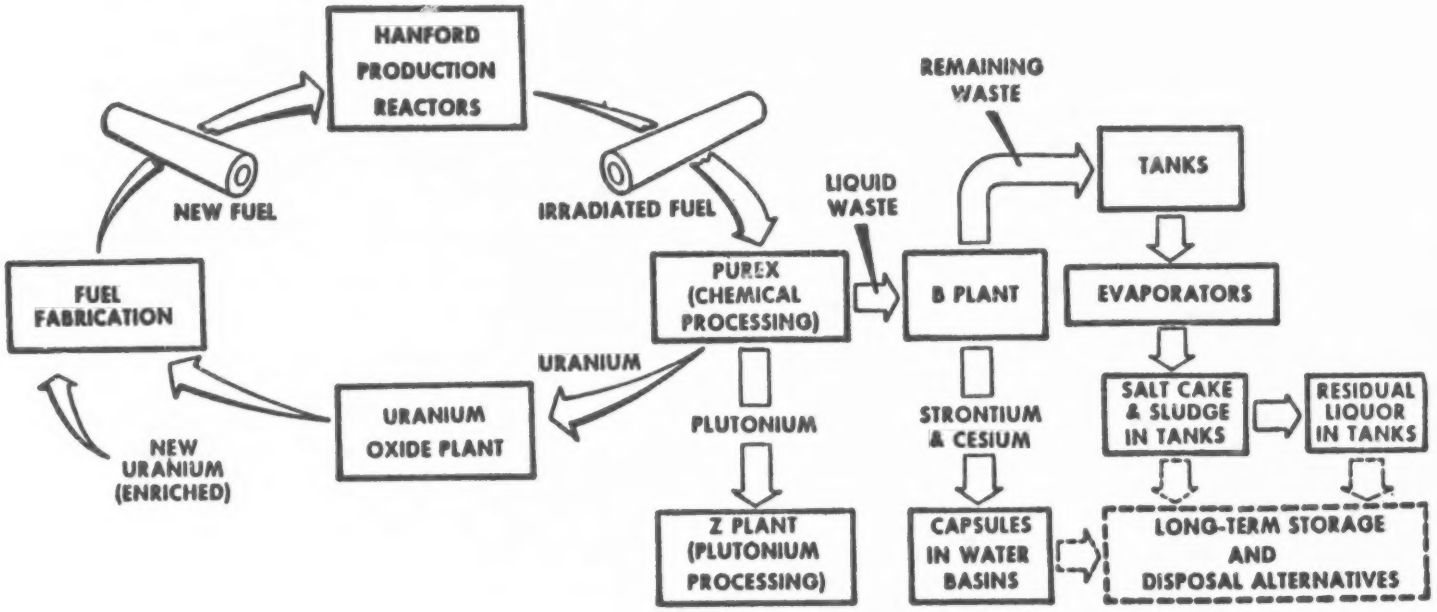

Figure 2.13. Hanford radioactive waste management.[37]

The high level liquid waste at Hanford was generated by the Purex, Redox, Thorex, Bismuth Phosphate ($BiPO_4$) and Tributylphosphate (TBP) processes. These chemical operations produced liquid wastes which differ significantly from each other. The Redox plant was shut down in 1966 and the last Purex plant campaign was completed in 1972. Thus, no high heat wastes are being generated at Hanford at the present time.[38] The Purex plant is scheduled to start up again to process the accumulated N-reactor fuel. The reactor fuel processing and waste management facilities are located in the 200 Areas. The sources and the types of radioactive wastes generated at Hanford are shown in Table 2.49. Both currently generated and self boiling wastes are processed by the B plant, which removes Sr-90 and Cs-137 from the high level radioactive waste resulting from the processing of Hanford reactor fuels. Following recovery in the B plant, the cesium and strontium are converted to CsCl and SrF_2 in the Waste Encapsulation and Storage Facility (WESF).

2.2.3.1 High Level Waste Inventory

High level radioactive waste generated from the processing of the Hanford reactor fuels is stored in large underground tanks. A summary of the tank farm is given in Table 2.50[36] which shows the number of tanks, the capacity of each tank, the year of construction and the cost. There is a total of 156 tanks in the 200 East and 200 West tanks farms. This number will soon be 168 when the 12 new double shell tanks are completed. Of the total 149 single tanks, 106 at present have been removed from active liquid service.[37] All these single shell tanks will be removed from active liquid service by January 1, 1981, and all pumpable liquids will be stored in the double shell tanks by the early 1980's. The present total capacity of the tank farm is 101 million gallons. A description of the liquid and solid waste inventory at Hanford is shown in Figure 2.14 as a function of time from 1945-1985.[40] Total volumes and masses of existing and future Purex wastes of salt cake, sludge, slurry and capsules are given in Table 2.51. The total volume of existing high level wastes at Hanford is then about 190,000 m^3 or 50 million gallons. The masses of the major fission product radionuclides in the Hanford tanks are given in Table 2.52 as a function of time after 1990.[40] It is found that there is only about six metric tons of existing fission products at Hanford. Masses of waste elements are given in Table 2.53.[41]

Estimated[37] curies of existing major fission product isotopes in the high level wastes are shown in Table 2.54. These were generated in Hanford production reactors from 1944-1971. Table 2.54 estimates have been decayed to 1990. Another estimate of the activity of in-tank wastes has been recently obtained from Hanford,[40] and is shown in Table 2.55 for several decay times after 1990. The activities in these two tables for Sr-90, Zr-93, and Sm-151 are similar but that of Cs-137, Tc-99 and Pm-147 are rather different. These discrepancies need to be resolved.

Table 2.49

Radioactive Waste Sources at Hanford[36]

Source	Type of Waste					
	Liquid					
	High Level		Intermediate Level	Low Level	Solid	Gaseous
	Boiling	Non-Boiling				
200 EAST AREA						
Purex Plant	X	X	X	X	X	X
B Plant	X	X	X	X	X	X
244-AR Vault	-	-	X	X	X	X
241-A, AX, AY Tank Farms	-	-	X	X	X	X
Critical Mass Laboratory	-	-	-	X	X	X
Semiworks	-	-	-	-	-	X
CR Vault	-	-	-	-	-	X
ITS-1 and 2 Concentrators	-	-	X	X	X	X
241-BX, BY and C Tank Farms	-	-	-	-	-	X
200 WEST AREA						
T Plant	-	X	X	X	X	X
242-T Concentrator	-	-	X	X	X	X
Laundry	-	-	-	X	X	X
231-Z	-	-	-	X	X	X
U Plant	-	-	-	X	-	X
Z Plant	-	X	X	X	X	X
UO_3 Plant	-	-	X	X	X	X
222-S Laboratory	-	X	X	X	X	X
241-SX Tank Farm	-	-	-	X	-	X
Redox Plant	-	-	-	-	-	X
241 TX Tank Farm	-	-	-	-	-	X
242-S Evaporator	-	-	-	X	X	X
FROM OTHER CONTRACTORS						
Pacific Northwest Laboratory	-	X	X	X	X	X
United Nuclear Industries, Inc.	-	X	X	X	X	X
Westinghouse Hanford Company	-	-	X	X	X	X
J. A. Jones Construction Company	-	-	-	-	X	-

Table 2.50

Hanford High Level Waste Storage Tanks[36,37]

	Tank Farm	No. of Tanks[a]	Capacity per Tank (10^3 gal)	Construction Type	Capacity per Farm (10^3 gal)	Year	Cost per Farm[b] (10^3 $)	Cost per gallon ($)
200 East Tank Farms (70 Tanks)	A	6	1000	single shell	6000	1954-55	5865	0.978
	AX	4	1000	single shell	4000	1963-64	2577	0.644
	B	12	530	single shell	6578	1943-44	3019	0.459
	B	4	55	single shell				
	BX	12	530	single shell	6360	1947-48	2208	0.347
	BY	12	758	single shell	9096	1950-51	2651	0.291
	C	12	530	single shell	6578	1943-44	2938	0.447
	C	4	55	single shell				
	AY	2	1000	double shell	2000	1968-70	3125	1.562
	AZ	2	1000	double shell	2000	1971-75	6200	3.100
200 West Tank Farms (86 Tanks)	S	12	758	single shell	9096	1950-51	3961	0.435
	SX	15	1000	single shell	15000	1953-54	3983	0.266
	SY	3	1000	double shell	3000	1975-77	--	--
	T	12	530	single shell	6578	--	3087	0.469
	T	4	55	single shell				
	TX	18	758	single shell	13644	1947-48	5859	0.429
	TY	6	758	single shell	4548	1951-52	1846	0.406
	U	12	530	single shell	6578	1943-44	2969	0.451
	U	4	55	single shell				
TOTAL		156			101056			

[a]12 more double shell 1,000,000-gallon capacity tanks (6 for Tank Farm AW and 6 for Tank Farm AN) are under construction in the 200 East Tank Farms. This will bring the total number of underground storage tanks to 168. Construction of one more double shell tank is under consideration.

[b]Includes original tank farm cost plus improvements, including instrumentation, agitation systems, and transfer lines from separations plants. Operating costs are not included.

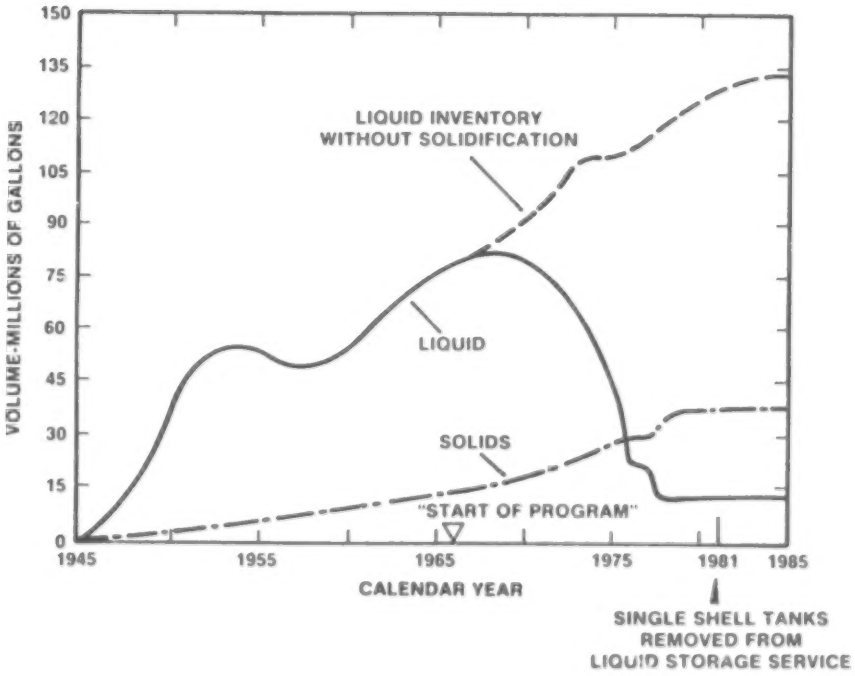

Figure 2.14. Volumes of high level waste at Hanford from 1945-1985.[40]

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Table 2.51

Estimates of HLW Quantities at Hanford[40]

Waste Categories	Existing Waste		Future Purex Waste		Totals	
	m^3	t*	m^3	t	m^3	t
Tank Waste						
Damp Salt Cake	95,000	136,000	1,000	1,600	96,000	138,000
Damp Sludge	49,000	77,000	8,000	9,400	57,000	87,000
Residual Slurry						
Double Shell Slurry	35,000	52,300	3,000	5,400	38,000	58,000
Complexant Concentrate	11,000	16,600	2,000	3,200	13,000	19,000
Capsule Waste						
CsCl Capsules**	3.5	5.3	1.4	2.2	4.9	7.5
SrF_2 Capsules†	1.0	1.6	0.75	1.2	1.8	2.8

* t = metric ton (1000 kg) (1 m^3 = 264.2 gallons).
**Existing cesium will produce 1910 capsules and future cesium 770 capsules.
†Existing strontium will produce 560 capsules and future strontium 420 capsules.

Table 2.52

Total Mass (kg) of In-Tank Existing High Level Fission Products at Hanford as a Function of Time After 1990*[40]

Radionuclide	Years After 1990					
	5	20	10^2	10^3	10^4	10^6
Se-79	11.7	11.7	11.7	11.6	10.5	2.75 E-1
Sr-90	392	271	37.7	8.7 E-9	0.	0.
Zr-90	51.4	173.	406	444	444	444
Zr-93	2680	2680	2680	2680	2670	1690
Nb-93	7.25 E-4	9.22 E-3	9.96 E-2	1.21	12.3	991
Ru-99	2.9 E-2	1.16 E-1	5.83 E-1	5.81	57.3	1710
Tc-99	1780	1780	1780	1780	1720	67.9
Pd-107	119	119	119	119	119	108
I-129	286	286	286	286	285	275
Cs-135	215	215	215	215	214	170
Pa-137	27.3	92.7	226	251	251	251
Sm-151	50.9	45.2	23.9	1.85 E-2	0.	0.

Table 2.52 Continued

Subtotal	5836.3	5831.7	5810.9	5793.6	5783.1	5707.2
Total**	5840	5840	5840	5840	5840	5840
Total including future additions	7120	7120	7120	7120	7120	7120

* This table does not include stable products. Also note that major portions of strontium and cesium are removed from the high-level liquid waste.
**The total amount includes other radioactive fission and decay products.

Table 2.53

Mass of Exisiting In-Tank Waste Elements in Hanford as of 1995*

Elements	Mass Metric Ton	Elements	Mass Metric Ton	Elements	Mass Metric Ton
Se	0.1	Sn	0.05	Th	15.0
Br	0.05	Te	0.7		
Rb	0.9	I	0.3	U-235	5.9
Sr	1.0	Cs	0.9	U-238	900.0
Y	1.2	Ba	3.9	Np	0.2
Zr	8.8	La	2.6	Pu-239	0.35
Mo	6.8	Ce	5.0		
Tc	1.8	Pr	2.4	Others	0.2
Ru	3.4	Nd	8.6		
Rh	1.0	Sm	1.8		
Pd	0.8	Eu	0.9		
Subtotal	25.85		27.15		921.65

Grand Total 974.65 metric tons

*The amounts include both radioactive and stable elements.

Table 2.54

Inventory of Major Fission Products in Hanford High Level Waste
Decayed to 1990
(From Reactor Production 1944-1971) in Curies[37]

Radionuclide[a]	Salt Cake[b]	Sludge	Residual Liquor	Capsules	Total
H-3	*	*	1.1×10^{4}	-	1.1×10^{4}
C-14	*	*	*		$<1.6 \times 10^{2}$
Se-79	*	8.2×10^{2}	*	-	8.2×10^{8}
Sr-90	2.0×10^{6}	4.5×10^{7}	6.0×10^{5}	5.8×10^{7}	1.1×10^{8}
Zr-93	*	6.9×10^{3}	*	-	6.9×10^{3}
Tc-99	*	*	3.1×10^{4}	-	3.1×10^{4}
Ru-106	*	3.7×10^{1}	3.7×10^{1}	-	7.4×10^{1}
Pd-107	*	5.7×10^{1}	*	-	5.7×10^{1}
Cd-113m	*	5.0×10^{3}	*	-	5.0×10^{3}
Sn-121m	*	1.1×10^{2}	*	-	1.1×10^{2}
Sn-126	*	6.8×10^{1}	*	-	6.8×10^{1}
Sb-125	*	2.0×10^{4}	*	-	2.0×10^{4}
I-129	*	*	4.7×10^{1}	-	4.7×10^{1}
Cs-134	1.2×10^{2}	1.2×10^{2}	4.3×10^{2}	2.5×10^{3}	3.2×10^{3}
Cs-135	4.4×10^{1}	4.4×10^{6}	1.6×10^{2}	9.0×10^{2}	1.1×10^{8}
Cs-137	5.0×10^{6}	5.0×10^{6}	1.8×10^{7}	1.0×10^{8}	1.3×10^{8}
Ce-144	*	9.9×10^{0}	*	-	9.9×10^{0}
Pm-147	*	1.0×10^{6}	*	-	1.0×10^{6}
Sm-151	*	1.4×10^{6}	*	-	1.4×10^{6}
Eu-152	*	1.5×10^{3}	*	-	1.5×10^{3}
Eu-154	*	7.3×10^{4}	*	-	7.3×10^{4}
Eu-155	*	7.4×10^{4}	*	-	7.4×10^{4}
TOTAL	7×10^{6}	52.6×10^{6}	18.6×10^{6}	158×10^{6}	243×10^{6}

[a]Daughter nuclides in decay chains are not listed. Curie values are for parent nuclide only.
[b]Radionuclides present in trace quantities are indicated by an asterisk.

Table 2.55

Total Activity (in millions curies) of In-Tank Existing High Level Fission Products at Hanford as a Function of Time After 1990

Radionuclide	Years After 1990					
	5	20	10^2	10^3	10^4	10^6
Sr-90	55.5	38.3	5.33	1.2 E-9	0	0
Y-90	55.5	38.3	5.33	1.2 E-9	0	0
Zr-93	6.87 E-3	6.87 E-3	6.87 E-3	6.86 E-3	6.84 E-3	4.33 E-3
Nb-93M	1.54 E-3	4.39 E-3	6.82 E-3	6.86 E-3	6.84 E-3	4.33 E-3
Tc-99	3.04 E-2	3.04 E-2	3.04 E-2	3.03 E-2	2.94 E-2	1.16 E-3
Cs-137	19.4	13.7	2.17	2.04 E-9	0	0
Ba-137M	18.1	12.9	2.03	1.91 E-9	0	0
Pm-147	0.27	5.05 E-2	3.28 E-9	0	0	0
Sm-151	1.39	1.23	0.65	5.02 E-4	0	0
Subtotal	150	104.4	15.5	0.045	0.04	9.82 E-3
Total	150	105	15.6	0.05	0.04	0.01
Total including future additions	168	109	16.2	0.06	0.05	0.01

Table 2.56 shows the mass, activity, and heating rates for the major actinides.[40] Table 2.57 gives further gross inventory and characteristics of solids stored in tanks at Hanford. There are 38 million gallons of salt cake in 73 tanks.[36] The Cs-137, Sr-90, and Pu-239 radionuclide contents in these wastes are given in Table 2.57 in Ci/gal (Reference 36) and Ci/L (Reference 38). There are 50 Ci/gal of Sr-90 in the SX Farm high heat sludges. The other radionuclides are much smaller in quantity. The radionuclide contents of the salt cake, however, differ considerably (by a factor of 2 to 4) between the results of Reference 36 and those of Reference 38. These wastes contain approximately 865 metric tons of uranium, 15 metric tons of thorium, 380 kg of plutonium and 7.5 kg of neptunium.[36] Much smaller quantities of americum and curium are present in these tanks. Table 2.58 shows the heating rates of high level fission products at Hanford tanks as a function of time after 1990.

The NUREG-0527 document[13] reports the following inventory for Hanford: Volume of high level waste in the tank farms is 47 million gallons and the activity is 108 x 10^6 curies; volume of Sr-90 and Cs-137 solutions held in storage in the B plant is approximately 28,000 gallons and the activity is 150 x 10^6 curies; the number of Sr-90 and Cs-137 capsules held at the Waste Encapsulation and Storage Facility is 400 and the activity is 31 x 10^6 curies. According to Reference 13, then, 289 x 10^6 curies of radioactive high level

Table 2.56

Existing Total Mass (kg), Activity (Ci) and Thermal Power (w) of In-Tank Actinides at Hanford as a Function of Time After 1990[40]

Radio-nuclide	Mass (kg) Years After 1990			Activity (Ci) Years After 1990			Thermal Power (W) Years After 1990		
	5	10	10	5	10	10	5	10	10
Th-232	1.47 E 4	1.47 E 4	1.47 E 4	1.61	1.61	1.61	3.90 E-2	3.90 E-2	3.90 E-2
U-233	4.19 E 1	4.18 E 1	9.89	397	396	93.8	1.16 E 1	1.15 E 1	2.73
U-234	6.69 E 1	6.69 E 1	4.96 E 1	414	414	307	1.19 E 1	1.19 E 1	8.83
U-235	5.88 E 3	5.88 E 3	6.21 E 3	12.6	12.6	13.3	3.49 E-1	3.50 E-1	0.37
U-238	9.00 E 5	9.00 E 5	9.00 E 5	300	300	300	7.58	7.58	7.58
Np-237	1.50 E 2	1.60 E 2	1.18 E 2	106	113	83.2	3.10	3.32	2.44
Pu-239	3.47 E 2	3.37 E 2	1.62 E-4	2.13 E 4	2.07 E 4	9.95 E-9	6.60 E 2	6.42 E 2	3.09 E-10
Pu-240	2.38 E 1	2.15 E 1	0.00	5.24 E 3	4.73 E 3	0.0	1.63 E 2	1.47 E 2	0.0
Pu-241	4.98 E-1	0.0	0.0	4.99 E 4	0.0	0.0	2.07	0.0	0.0
Am-241	1.31 E 1	2.76	0.00	4.48 E 4	9.47 E 3	0.0	1.49 E 3	3.16 E 2	0.0
Cm-244	1.43 E-2	4.14 E-13	0.00	1.16 E 3	0.0	0.0	4.06 E 1	0.0	0.0
Subtotal	9.21 E 5	9.21 E 5	9.21 E 5	1.24 E 5	3.61 E 4	7.99 E 2			
Total	9.21 E 5	9.21 E 5	9.21 E 5	1.26 E 5	3.74 E 4	5.60 E 3	2.42 E 3	1.15 E 3	1.17 E 2

Table 2.57

Estimated Inventory and Characteristics of Solids Stored in Tanks Year 1980[36,38]

	Salt Cake	High Heat Sludge (SX Farms)	Other Sludges
Gallons (millions)	38[a]	0.6	8
Number of Tanks	73	7	72
Radionuclides (Ci/gal)[29]			
Cs-137	0.6	5.0	0.3
Sr-90	3×10^{-2}	50	2
Pu-239	4×10^{-4}	4×10^{-3}	6×10^{-4}
Radionuclides (Ci/liter)[31]			
Cs-137	0.04		
Sr-90	0.004		
Pu-239	$<10^{-4}$		

[a] 38×10^6 gallons = 143.8×10^6 liters.
Estimated total activity in salt cake:
approximately 22.9×10^6 Curies from Reference 36 data;
approximately 6.3×10^6 Curies from Reference 38 data.

Table 2.58

Total Thermal Power (kw) of In-Tank Existing High Level Fission Product Wastes at Hanford as a Function of Time After 1990

	Years After 1990						
Radio-nuclide	5	20	100	200	250	300	1000
Sr-90	72.4	50.2	6.99	0.59	0.17	0.05	1.6 E-9
Y-90	327	226	31.4	2.67	0.78	0.23	7.3 E-9
Zr-93	8.14 E-4	8.14 E-4	8.14 E-4	8.14 E-4	8.14 E-4	8.14 E-4	8.14 E-4
Nb-93m	2.75 E-4	7.8 E-4	1.21 E-3	1.22 E-3	1.22 E-3	1.22 E-2	1.22 E-3
Tc-99	2.05 E-2	2.05 E-2	2.05 E-2	2.05 E-2	2.05 E-2	2.05 E-2	2.05 E-2
Cs-137	31.8	22.5	3.55	0.35	0.11	3.50 E-2	3.34 E-9
Ba-137m	71.4	50.5	7.96	0.79	0.25	7.86 E-2	7.50 E-9
Sm-151	2.42	2.14	1.13	0.51	0.34	0.23	8.76 E-4
Total	506	351	51.1	4.94	1.68	0.65	0.025
Total including future additions	541	364	52.9	5.15	1.77	0.69	.031

waste is stored at Hanford. The IRG Report[6] gives approximately 12 million gallons of liquid waste, approximately 21 million gallons of salt cake, and approximately 13 million gallons of sludge for a total of 46 million gallons of high level waste as of October 1977.

Figures 2.15 and 2.16 show the time dependent radionuclide distribution in curies of the in-tank and encapsulated wastes, respectively, for 1 million years after 1990. The radioisotopes with less than one curie of inventory are not shown in these tables and figures. The daughter nuclides of the radioisotopes are also not shown in these estimates. As shown in Figure 2.15, after 1000 years, the radioisotopes that will be the controlling isotopes are Tc-99, Pu-239, Am-241, Zr-93, and Pu-240. Similarly, the Cs-137, Sr-90, and their daughters Y-90 and Ba-137 will die out in 1000 years in the encapsulated waste and Cs-135 will be the only major radioisotope in these wastes.

2.2.3.2 Chemical Composition of Waste

The average chemical composition of Hanford high level waste is shown in Table 2.59.[40] It is derived from a limited number of sample analyses and detailed records of fuel elements processed and chemicals used. It is known that the detailed compositions of the waste vary from tank to tank; the results of Table 2.59 may be considered a reasonable estimation of the Hanford waste.[37]

The salt cake is mostly $NaNO_3$ (84.5 wt %) and a mixture of $NaNO_2$, NaOH, $NaAlO_2$, and Na_2CO_3. It contains small amounts of Sr-90, Cs-137, Pu-239, and other radioisotopes. The residual liquor contains mostly $NaNO_3$, NaOH, $NaNO_2$, $NaAlO_2$, and water. The sludge has Na_3PO_4 and $NaNO_3$ as major compounds totaling about 36 wt % and many other compounds each contributing about 1 to 4 wt % as shown in Table 2.59. Table 2.60 gives the chemical compositions of salt cakes and sludges in SX and other farms from Reference 36. There are some obvious discrepancies between the two tables. As for example, $NaNO_3$ composition in sludges is about three to four times higher in Table 2.60 contents than what is reported in the average composition in Table 2.59. The salt composition range reported in Reference 38 agrees with the results of Table 2.60. Thus, detailed data from different tank farms will be needed to obtain an accurate chemical characterization of the Hanford waste.

The chemical concentrations of different elements in the high heat and low heat generating wastes are shown in Table 2.61 and 2.62, respectively.[38] The high heat generating liquid waste (primarily from the Purex and Redox processes) contains greater than 99% of the fission products from the irradiated uranium fuels. Low heat wastes contain relatively small quantities of fission products. These wastes, shown in Table 2.62, consist of stored $BiPO_4$ process and early Redox process waste, tributyl phosphate process wastes from an early uranium recovery program, process solvent wash wastes, and fuel cladding removal wastes. The uncertainties of the estimates given in Tables 2.60, 2.61, and 2.62 are not known to us.

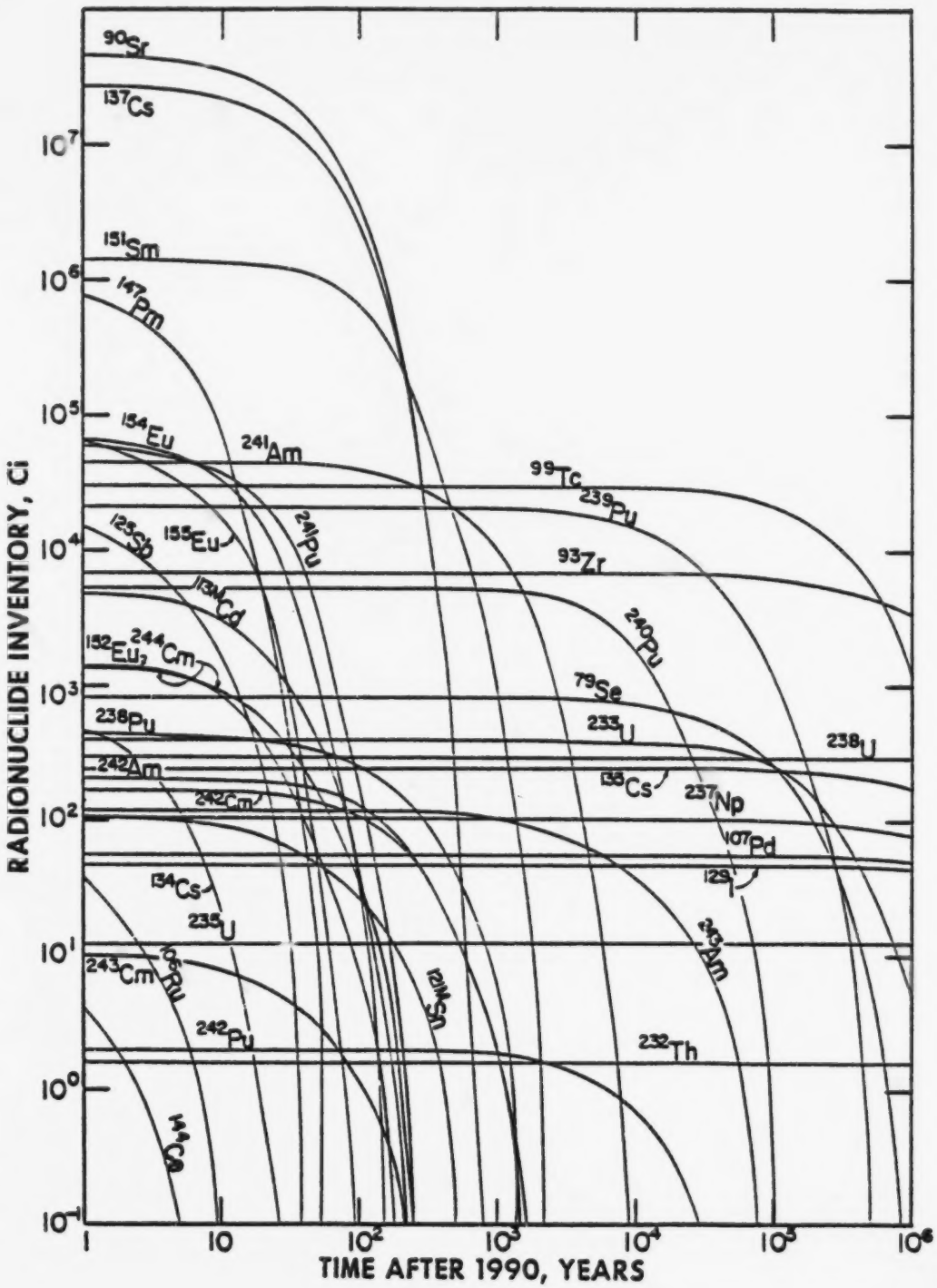

Figure 2.15. Radionuclide content of Hanford in-tank waste.[37]

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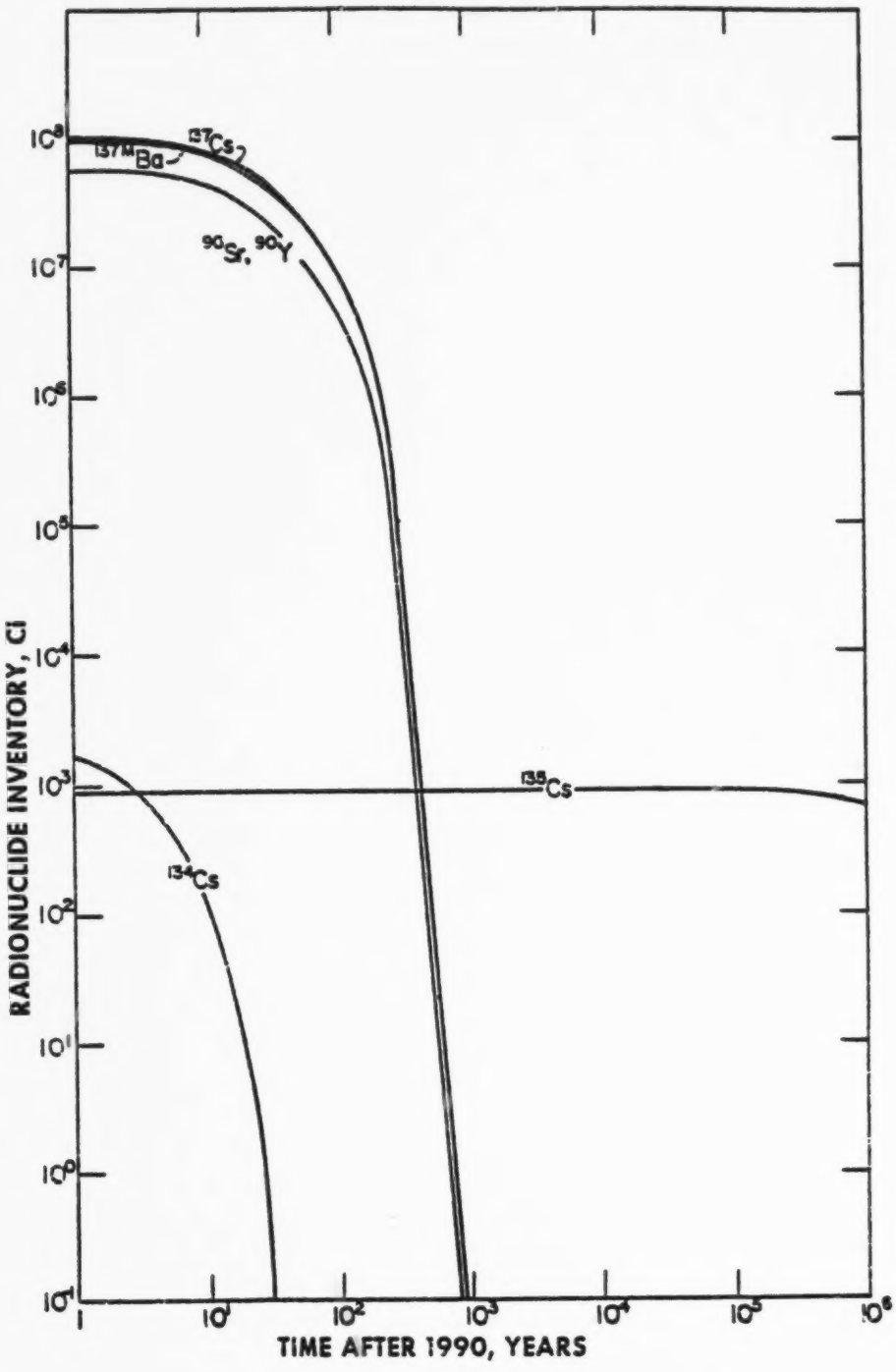

Figure 2.16. Radionuclide content of Hanford encapsulated waste.[37]

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Table 2.59

Major Chemicals in Hanford High Level Defense Waste In-Tank Inventory*[40]

Chemical	Salt Cake Wt %	Sludge Wt %	Residual Slurry Wt %	Total Wt %
$NaNO_3$	84.5	16.3	15.5	46.7
H_2O	10.3	39.5	45.9	28.1
Na_3PO_4	0.1	19.9	----	5.9
NaOH	1.5	3.2	12.8	4.9
$NaNO_2$	1.7	3.3	10.8	4.5
$NaAlO_2$	1.4	2.7	10.6	4.2
Na_2CO_3	0.5	2.0	1.3	1.1
Na_2SiO_3	----	2.5	----	0.7
Al_2O_3	----	2.0	----	0.6
$Fe(OH)_3$	----	1.6	----	0.5
Na_2SO_4	----	1.0	1.0	0.6
NaF	----	1.1	0.5	0.4
ZrO_2 $2H_2O$	----	0.8	----	0.2
KF	----	0.2	0.1	0.1
KOH	----	0.3	0.1	0.1
Total Organic Carbon	----	----	0.3	0.1
Other	----	3.6	1.1	1.3
TOTALS	100.0	100.0	100.0	100.0

*Percentage values are calculated to up to three places (rounded to nearest 0.1%) for purposes of totaling 100%. Three- or even two-place accuracy on the values is not implied. Trace amounts (less than 0.1%) of material may be present in waste form where (----) appears. Compositions are averaged over existing plus future waste categories.

Table 2.60

Average Compositions of Solids Stored in Tanks at Hanford (1980)[36]

Chemical	Salt Cake In Tank Solidified Waste Wt. %	Salt Cake Neutralized Waste Wt. %	High Heat Sludge (SX Farm) Wt. %	Other Sludges Wt. %
$NaNO_3$	80	20	45	75
$NaNO_2$	4	1	7	4
NaOH	10	1	5	1
$NaAlO_2$	1	2	20	--
Na_2CO_3	2	1	10	--
Other (Fe, SO_4, PO_4)	1	25	3	18
Water of Hydration	2	50	10	2
TOTAL	100.0	100.0	100.0	100.0
Bulk Density (g/cc)	1.8	1.8	2.0	2.0

Table 2.61

Concentrations of Hanford High Heat Wastes (in Molarity)[38]

Constituent	Current Acid Wastes PAW[a]	Current Acid Wastes ZAW[b]	Stored Neutralized Wastes Purex	Stored Neutralized Wastes Redox
H	1.0	1.0	---	---
Na	0.7	0.4	3.5	7.0
Fe	0.25	0.26	0.35	---
Al	0.25	0.35	---	---
Zr	---	0.03	---	---
AlO_2	---	---	0.2	1.2
NO_2-NO_3	2.4	2.1	3.1	3.6
CO_3	---	---	---	0.1
SO_4	0.4	0.6	0.5	0.03
PO_4	0.01	0.01	0.01	---
OH	---	---	pH 9-11	pH 7.0
F	0.005	0.1	0.004	---
SiO_3	0.1	0.01	0.1	---

[a]From Al-clad fuel.
[b]From Zr-clad fuel.

Table 2.62

Concentrations of Hanford Low Heat Wastes (in Molarity)[38]

Constituent	Neutralized		Coating Waste		Purex Organic
	$BiPO_4$	TBP	Al-Clad	Zr-Clad	Wash Waste
Na	7.0	8.0	5.0	3.0	0.15
Fe	0.1	0.06	---	---	---
Bi	0.03	---	---	---	---
Zr	---	---	---	0.4	---
AlO_2	---	---	2.0	---	---
MnO_2	---	---	---	---	0.02
NO_2-NO_3	5.3	6.6	1.7	1.7	0.1
CO_3	---	---	---	---	0.06
SO_4	0.16	0.31	0.03	---	---
PO_4	0.9	0.29	---	---	---
OH	---	0.18	1.2	---	0.03
F	0	0	0	2.9	---

Sutherland and Bennett[20] have computed the radionuclide content of the waste form if the Hanford defense high level waste were solidified in glass form. This is shown in Table 2.63. The curies and heating rates of this waste are shown in Table 2.64 as a function of time from 1 to 10^5 years. The Hanford waste as given in Table 2.63 and shown in Figure 2.4, has lower heating rates during the first 200 years when compared to the SRP and INEL wastes because the Sr and Cs have been removed from this waste.

Table 2.63

Radionuclide Content of Hanford Defense High Level Waste in Glass Form

Isotope	Curies/m^3*	Isotope	Curies/m^3
C-14	4.40 - 1	Ce-144	7.90 + 2
Ni-59	3.35 - 2	Pm-147	1.85 + 4
Ni-63	2.21 + 1	Sm-151	3.43 + 2
Co-60	5.23 + 1	Eu-152	1.55 + 0
Se-79	4.85 - 1	Eu-154	4.15 + 2
Sr-90	7.50 + 3	Eu-155	3.95 + 1
Zr-93	4.00 + 0	U-238	9.50 - 3
Tc-99	1.81 + 1	Np-237	7.00 - 2
Ru-106	5.48 + 2	Pu-238	1.18 + 0
Pd-107	5.60 - 2	Pu-239	6.93 + 0
Ag-110m	4.15 - 1	Pu-240	3.48 + 0
Cd-113m	5.95 + 0	Pu-241	3.18 + 2

Table 2.63 Continued

Isotope	Curies/m^3*	Isotope	Curies/m^3
Sn-119m	2.68 - 2	Am-241	3.15 + 2
Sn-121m	1.38 - 1	Am-242m	2.53 + 0
Sn-126	4.70 - 2	Am-243	1.72 + 0
Sb-125	3.99 + 2	Cm-242	2.07 + 0
I-129	3.03 - 2	Cm-243	2.01 - 1
Cs-134	2.17 + 1	Cm-244	2.98 + 1
Cs-135	2.85 - 2	Cm-245	3.00 - 3
Cs-137	5.05 + 3		

*Ci/m^3 of the final product.

Table 2.64

Hanford Defense High Level Waste Activity and Heating Rate

Waste Age (yr)	Curies/m^3	Watts/m^3	Waste Age (yr)	Curies/m^3	Watts/m^3
1	4.10 + 4	1.03 + 2	1000	1.09 + 2	2.70 + 0
2	3.65 + 4	9.60 + 1	2000	5.29 + 1	8.35 - 1
3	3.31 + 4	9.15 + 1	3000	4.11 + 1	4.45 - 1
4	3.05 + 4	8.79 + 1	4000	3.81 + 1	3.53 - 1
5	2.84 + 4	8.50 + 1	5000	3.70 + 1	3.23 - 1
6	2.66 + 4	8.24 + 1	6000	3.63 + 1	3.05 - 1
7	2.52 + 4	8.01 + 1	7000	3.57 + 1	2.91 - 1
8	2.40 + 4	7.79 + 1	8000	3.52 + 1	2.79 - 1
9	2.29 + 4	7.59 + 1	9000	3.47 + 1	2.67 - 1
10	2.20 + 4	7.40 + 1	10000	3.43 + 1	2.56 - 1
20	1.64 + 4	5.90 + 1	20000	3.07 + 1	1.77 - 1
30	1.28 + 4	4.82 + 1	30000	2.84 + 1	1.31 - 1
40	1.02 + 4	3.99 + 1	40000	2.68 + 1	1.01 - 1
50	8.09 + 3	3.35 + 1	50000	2.56 + 1	8.12 - 2
60	6.47 + 3	2.85 + 1	60000	2.46 + 1	6.69 - 2
70	5.18 + 3	2.45 + 1	70000	2.37 + 1	5.65 - 2
80	4.18 + 3	2.14 + 1	80000	2.29 + 1	4.90 - 2
90	3.38 + 3	1.89 + 1	90000	2.22 + 1	4.35 - 2
100	2.75 + 3	1.69 + 1	100000	2.16 + 1	3.95 - 2
200	5.76 + 2	9.40 + 0			
300	3.11 + 2	7.59 + 0			
400	2.42 + 2	6.48 + 0			
500	2.04 + 2	5.56 + 0			
600	1.76 + 2	4.79 + 0			
700	1.54 + 2	4.14 + 0			
800	1.36 + 2	3.58 + 0			
900	1.22 + 2	3.10 + 0			

2.2.3.3 Summary and Projection of the Hanford High Level Waste

The present program and projections of the Hanford high level wastes have been summarized in the ERDA-77-44 document.[37] The following is a synopsis of the high level waste projections at Hanford.

As shown in Table 2.50, the waste storage system in the early 1980's will consist of 149 single shell tanks for salt cake and sludge, 20 double shell tanks for residual liquor and other liquid waste solutions, and 10 water basins for strontium and cesium capsules. At that time, the high level waste inventory will consist of approximately:[37]

- 25 million gallons (bulk) of damp salt cake
- 11 million gallons (bulk) of damp sludge
- 3 million gallons of liquid wastes awaiting solidification
- 11 million gallons of residual liquor
- 2,900 capsules containing Sr-90 or Cs-137 as strontium fluoride and cesium chloride.

The quantity of the salt cake may increase if the residual liquor is converted to a solid by techniques under study at present. Note that the above estimates do not include N-reactor fuel that will be reprocessed by future Purex operation. The above estimates agree with the Savannah River strategy document[18] which states that 105 million gallons of high level waste were generated at Hanford and are concentrated to about 50 million gallons (1978). According to their estimates this may become 65 million gallons in 1990.

The salt cake, mostly sodium nitrates, will stay as layers up to 25-feet thick in approximately one-third of the single shell tanks. The salt cake crystals will contain about 5 million gallons of interstitial liquid within its total volume. Each salt cake tank may contain a heel of approximately 30,000 gallons of drainable but non-pumpable liquor.[37] The salt cake is estimated to have 7×10^6 curies of fission products (decayed to 1990), and a small fraction of the estimated 400 kg of total plutonium in the waste tanks.[37]

The sludge consists of insoluble metal oxides and hydroxides in layers from a few inches thick up to 15-feet thick at the bottoms of the most underground tanks. Fifty percent of the sludge is sludge solids and the rest non-pumpable interstitial liquid. The sludge is estimated to have about 50×10^6 curies of fission products (decayed to 1990) and most of the plutonium (approximately 8×10^{-3} Ci/gal).

The residual liquor is caustic liquid which could not be converted to solid by simple evaporation. This will be stored in double shell tanks. It will have approximately 20×10^6 curies of fission products (decayed to 1990), and about one kg of plutonium.[37] Work is under way at Hanford to develop a process to reduce the volume of this residual liquor.

It is estimated that the Hanford N-reactor has accumulated a backlog of about 2026 MT of spent fuel[39] (see Section 3 below). It will produce additional amounts at the rate of 400-900 MT/yr.[39] It is estimated that if the Purex plant is operating,

approximately 1 million gallons of liquid HLW will be generated at Hanford each year.

The total waste inventory at Hanford including the processing of the N-reactor fuels generated after 1972 is given in Table 2.65. It shows that the N-reactor fuels will increase the in-tank waste volumes in 1990 by about 5%, and will increase the number of capsules by 40%. It will also increase the long half life (>25 years) fission products inventory by about 16%. The addition of the new waste to the old waste will relatively increase the short lived fission products (such as Ru-106), and this will have to be considered in the waste solidification scheme.

The Savannah River document[18] states that there were 190 x 10^6 curies of alkaline salt, sludge and liquid and 350 x 10^6 curies of Sr-90/Y and Cs-137/Ba wastes at Hanford in 1978. According to this report, these wastes may increase to 230 x 10^6 and 425 x 10^6 curies, respectively, in 1990. Thus, there may be as high as 655 megacuries of radioactive high level waste at Hanford in 1990.

Table 2.65

Estimated Hanford High Level Waste in 1990[37,42]

	Salt Cake (10^6 gal)	Sludge (10^6 gal)	Residual Liquor (10^6 gal)	Number of Capsules Sr	Cs
Projections From Hanford Reactor Operations (1944-1971)*	25 [24.3]	11 [12.3]	11 [11.9]	600	2300
From N-Reactor Operations (1972-1990)	1.3	0.2	1	∿410	∿670
TOTAL	26.3	11.2	12	∿1010	∿2970
TOTAL from Reference 42	25.1	15.1	13.7		

*Numbers inside brackets are from Reference 42, and indicate volumes as of December 31, 1979.

Blomeke at Oak Ridge[42] has determined the projected volumes of Hanford wastes based on the assumption that the N-reactor will operate through 1990 producing fuel-grade plutonium. These are given in Table 2.66 along with their projections of waste volumes for Savannah River and Idaho Falls.

Table 2.66

Projected Volumes of High Level Defense Wastes to be Accumulated (10^3 m^3)[42]

Calendar Year Ending	Liquid	Salt Cake	Sludge	Calcine	Cs, Sr Capsules	Total
			Savannah River			
1980	51.5	25.7	11.0	----	----	88.2
1985	31.8	44.7	13.2	----	----	89.7
1990	19.7	54.9	14.8	----	----	89.3
1995	18.9	61.7	17.0	----	----	97.7
2000	22.3	64.3	17.0	----	----	103.7
			Idaho			
1980	8.5	----	----	2.1	----	10.6
1985	4.7	----	----	3.5	----	8.2
1990	5.6	----	----	4.7	----	10.3
1995	4.6	----	----	6.7	----	11.3
2000	3.8	----	----	8.6	----	12.4
			Hanford[a]			
1980	39.0	95.0	49.0	----	0.0017	183.0
1985	50.0	95.0	51.0	----	0.0043	196.0
1990	52.0	95.0	57.0	----	0.0067	204.0
1995	52.0	95.0	57.0	----	0.0067	204.0
2000	52.0	95.0	57.0	----	0.0067	204.0
			TOTAL			
1980	99.8	120.7	60.0	2.1	0.0017	281.8
1985	86.5	139.7	64.2	3.5	0.0043	293.9
1990	77.0	150.0	71.8	4.7	0.0067	303.6
1995	75.5	157.0	74.0	6.7	0.0067	313.0
2000	78.1	159.0	74.0	8.6	0.0067	320.1

[a]Projected volumes are highly sensitive to schedule for startup of the Purex plant.

2.3 Summary

The preceding four sections have reviewed the existing data available on high level radioactive waste inventories at the four principal sites. It is found there that the available data from several sources are sometimes not consistent with each other. There is also the additional problem that the activity values of the wastes are given after different times of decay and this has made it difficult to develop a consistent picture of the total high level radioactive waste in the U. S. Table 2.67 shows the gross amounts of high level waste stored at West Valley and at the three DOE facilities. This Table is mostly taken from the data given in Reference 18. This is the only consistent table available to us from which we can compare wastes at the different sites. The following observations can be made from this Table:

- Total volume of high level liquid, sludge, and salt cake in the U. S. A. was approximately 75 x 10^6 gallons in 1978 and may become 89 x 10^6 gallons in 1990.

- The percentages of the total volume of HLW (liquid, sludge, and salt cake) stored at the Hanford, Savannah River, and Idaho sites are 67%, 30%, and 3%, respectively, in 1978 and may become 74%, 25%, and approximately 1% in 1990 because of conversion of the liquid waste to solid forms.

- Total activity of high level waste consisting of the liquid, sludge, salt cake, capsules and calcines was 1.26 x 10^9 curies in 1978 and may become as high as 1.9 x 10^9 curies in 1990.

- The percentages of radioactivity in curies stored at the Hanford, Savannah River and the Idaho sites were 43%, 45%, and 7%, respectively, in 1978 and could change to 35%, 36%, and 27%, respectively, in 1990.

- A large number of spent fuel assemblies is stored at each site.

Some comments should now be made with respect to the data presented in this Table and in the preceding sections. First, the volume of waste projected by the Savannah River strategy document[18] differ somewhat from those projected by the Oak Ridge report.[42] As an example, the total volume of high level waste would be 89 x 10^6 gallons according to Reference 18 and should be 79 x 10^6 gallons according to Reference 42.

The neutralized Tank 8D2 in West Valley has 39 x 10^6 curies of total activity in 1978 as reported in Table 2.5. Reference 18 quotes this to be 64 x 10^6 curies in 1978. The activity of the acidic waste in Tank 8D4 at West Valley agrees reasonably well with the results of Reference 18. The amount of activity of the waste in Idaho is given at different times in Section 2.2.1, and cannot be compared with the results of Table 2.67. The actual reasons for the high estimate of 500 x 10^6 Ci of activity at Idaho in 1990 are not known to us at present, although this may be due to processing of different fuel types. NUREG-0527, Reference 13, estimates the radioactivity of the Savannah River tanks to be 775 x 10^6 curies, which is much higher compared to the number

Table 2.67

Summary of U.S. High Level Nuclear Wastes[18a]

		Hanford	Savannah River	Idaho	West Valley
No. of Tanks		168	33	15	4 (2 spare)
Total capacity: gal		113×10^6	b	3.4×10^6	0.76×10^6
Total salt, sludge and liquid: gal	(1978)	50×10^6	22×10^6[c]	$2.4\text{-}2.6 \times 10^6$	0.61×10^6
	(1990)	65×10^6[d]	22×10^6[e]	1.2×10^6[f]	0.61×10^6
Ci	(1978)	190×10^6	570×10^6	b	39×10^6[g]
	(1990)	230×10^6	680×10^6	b	50×10^6
Capsules: Ci	(1978)	350×10^6			
	(1990)	425×10^6			
Calcines: Ci				b	
Subtotal: Ci	(1978)	540×10^6	570×10^6	85×10^6	66×10^6
	(1990)	655×10^6	680×10^6	500×10^6	50×10^6
Amount of Pu in waste: kg		400	225	20 - 50	35
Spent Fuel		1800 MT + 400-900 MT/yr	b	6830 fuel elements plus naval reactor fuels	715 fuel assemblies plus 42 N reactor fuel elements

[a]1990 data are the high estimates obtained in Reference 18.
[b]Data unknown.
[c]Tank inventory is 23×10^6 gallons as of 8/20/79 as reported in Reference 29.
[d]According to Reference 42 this volume should be 53.9×10^6 gallons (see Table 2.66).
[e]According to Reference 42 this volume should be 23.6×10^6 gallons (see Table 2.66).
[f]According to Reference 42 this volume should be 1.48×10^6 gallons (see Table 2.66).
[g]Data from Reference 1.

shown in Table 2.67. This also needs to be resolved. The curies of radioactivity reported in Table 2.67 for Hanford, are on the other hand, higher than what has been found in NUREG-0527, which gives a total of only 289 x 10^6 curies of liquid and capsules at Hanford. Thus, there is a need for a careful investigation of the high level waste inventories at all these laboratories in order to arrive at a uniform picture.

The chemical compositions of the high level wastes at the different sites are not compared, because these should be examined on a site-specific basis and are dealt with extensively in the preceding sections.

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3. EXISTING COMMERCIAL SPENT FUEL INVENTORY

The existing spent fuel from commercial power reactors is stored primarily at the reactor sites in the spent fuel storage pools. Additionally, some spent fuel is stored at the General Electric, Morris, Illinois facility and at the Nuclear Fuel Services, West Valley site. The West Valley facility is no longer accepting spent fuel for storage.

Table 3.1 gives the inventory of exisiting commercial spent fuel as of December 31, 1979. The data on the number of fuel assemblies in storage were obtained from NUREG-0020[1] except for the NFS West Valley site. To express this in metric tons of heavy metal (MTHM) the data on core loading from the Directory of Nuclear Reactors[2] were consulted for each reactor. From these data the MTHM for each assembly was calculated. For some cases, as a spot check, the FASR for the specific reactor was consulted. There is a considerable variation in the size of the Westinghouse PWR assemblies. These assemblies contain between 255 and 521 MTHM per assembly.[3]

The spent fuel assemblies in storage at NFS West Valley are shown in Table 3.2. This Table is based on data supplied by Nuclear Fuel Services, Inc.[4] which was updated in a recent communication to BNL.[5] The total spent fuel stored at NFS from this Table is also included in Table 3.1 under Independent Spent Fuel Storage Installations. Table 3.3 gives the commercial fuel reprocessed at the NFS West Valley site. This is based on data supplied by NFS and discussed in Section 2 above.

The Three Mile Island Unit 2 fuel assemblies may also be considered spent fuel. These constitute about 82 MTHM. The total of the MTHM reprocessed at NFS West Valley and the spent fuel in storage (including the fuel from TMI-2) represents the existing (as of December 31, 1979) inventory of commercially generated spent fuel. Adding the total MTHM from Tables 3.1 and 3.3 and about 82 MTHM from TMI-2 gives about 5453 MTHM generated by the commercial power industry to December 1979. Of this amount generated, about 262 MTHM was reprocessed at NFS leaving about 5190 MTHM as spent fuel in storage. There are also some spent fuel assemblies stored at the DOE Idaho site. These are fuels from the Peach Bottom 1 and the Fermi 1 reactors. There are 1603 Peach Bottom 1 fuel elements and 510 Fermi 1 fuel elements stored at Idaho. Spent fuel is also stored at the Hanford N Reactor and the Fort St. Vrain reactor. The Hanford N Reactor is a dual purpose reactor located at the Hanford Reservation near Richland, Washington. As of September 30, 1979 there were 2026 metric tons of uranium contained in the spent fuel stored at the K reactor and N reactor storage pools.[6] This fuel may be reprocessed at the Hanford Purex Plant which is presently shut down but is scheduled for startup in a few years. There are 244 spent fuel elements stored at the Fort St. Vrain High Temperature Gas Cooled Reactor. These fuel elements contain a total of 2,829 kg of thorium-232, approximately 96 kg of uranium, and about 107 grams of plutonium.[7] This fuel is expected to be shipped to the Idaho National Engineering Laboratory sometime during 1980.

Table 3.1

Existing Commercial Spent Fuel Inventory as of 12/31/79[1]

Facility	Reactor Vendor	No. of Assemblies Stored	MTHM per Assembly (Approx.)[a]	Total MTHM Stored
Pressurized Water Reactors				
Arkansas 1	B&W	176	.465	81.8
Arkansas 2	CE	0	.465	
Beaver Valley 1	W	0	.445	
Calvert Cliffs 1	CE	364[b]	.375	136.5
Calvert Cliffs 2	CE		.375	
Cook 1	W	193	.445	85.9
Cook 2	W		.445	
Crystal River 3	B&W	60	.465	27.9
Davis-Besse 1	B&W	0	.465	
Farley 1	W	0	.445	
Fort Calhoun 1	CE	157	.375	58.9
Ginna	W	156	.390[c]	60.8
Haddam Neck	W	340	.410	139.4
Indian Point 1	B&W	160	.136	21.8
Indian Point 2	W	200	.445[c]	89.0
Indian Point 3	W	64	.445	28.5
Kewaunee	W	120	.340	40.8
Maine Yankee	CE	433	.375	162.4
Millstone 2	CE	144	.375[c]	54.0
North Anna 1	W	0	.390	
North Anna 2	W	0	.390	
Oconee 1	B&W	545[b]	.465	253.4
Oconee 2	B&W		.465	
Oconee 3	B&W		.465	
Palisades	CE	273	.450	122.9
Point Beach 1	W	212[b]	.390	82.7
Point Beach 2	W		.390	
Prairie Island 1	W	280[b]	.340	95.2
Prairie Island 2	W		.340	
Rancho Seco 1	B&W	112	.465	52.0
Robinson 2	W	120	.445	53.4
Salem 1	W	40	.390	15.6
San Onofre 1	W	58	.390	22.6
St. Lucie 1	CE	60	.390[c]	23.4
Surry 1	W	436[b]	.450	196.2
Surry 2	W		.450	
Three Mile Island 1	B&W	212	.465	98.6
Three Mile Island 2	B&W	0	.465	
Trojan	W	64	.410	26.2

Table 3.1 Continued

Facility	Reactor Vendor	No. of Assemblies Stored	MTHM per Assembly (Approx.)[a]	Total MTHM Stored
Turkey Point 3	W	346[b]	.390	134.9
Turkey Point 4	W		.390	
Yankee Rowe 1	W	149	.255	38.0
Zion 1	W	436[b]	.445[c]	194.0
Zion 2	W		.445[c]	
Boiling Water Reactors				
Big Rock Point 1	GE	88	.131	11.52
Browns Ferry 1	GE	576	.195	112.3
Browns Ferry 2	GE	400	.195	78.0
Browns Ferry 3	GE	354	.195	69.0
Brunswick 1	GE	154 PWR[d]	.445	68.5
		320 BWR	.195	62.4
Brunswick 2	GE	91 PWR[d]	.445	40.5
		132 BWR	.195	25.7
Cooper Station	GE	284	.195	55.4
Dresden 1	GE	221	.112	24.8
Dresden 2	GE	1069[b]	.195[c]	208.5
Dresden 3	GE		.195[c]	
Duane Arnold	GE	276	.195	53.8
Fitzpatrick	GE	268	.195	52.3
Hatch 1	GE	260	.185	48.1
Hatch 2	GE		.185	
Humboldt Bay	GE	251	.080	20.1
LaCrosse	AC	113	.106	12.0
Millstone 1	GE	776	.195	151.3
Monticello	GE	728	.195	142.0
Nine Mile Point 1	GE	660	.195	128.7
Oyster Creek 1	GE	620	.195	120.9
Peach Bottom 2	GE	618	.195	120.5
Peach Bottom 3	GE	712	.195	138.8
Pilgrim 1	GE	580	.195	113.1
Quad Cities 1	GE	596	.195[a]	116.2
Quad Cities 2	GE	492	.195[a]	95.9
Vermont Yankee 1	GE	894	.195	174.3

Table 3.1 Continued

Independent Spent Fuel Storage Installations

Morris Operations	GE	310.0[e]
West Valley	NFS	157.0

TOTALS

Total PWR Assemblies	Total BWR Assemblies	Total MTHM
6155	11,288	5108.4

[a]Based on calculations from data supplied in the Director of Nuclear Reactors, IAEA.
[b]Total stored for this and unit(s) following at this station.
[c]Estimate of MTHM per assembly (at loading) obtained from the FSAR.
[d]PWR assemblies are from Robinson 2.
[e]1 MTU = 2 PWR assemblies or 5 BWR assemblies.

AC - Allis-Chalmers
B&W - Babcock and Wilcox
CE - Combusion Engineering
GE - General Electric
NFS - Nuclear Fuel Services
W - Westinghouse

Table 3.2

NFS West Valley

Source and Quantity of Fuel Pool Contents
(Originally of 5/31/76; Updated 2/26/80)

Reactor	No. of Assemblies Stored	MTU per Assembly (Approx.)	Total MTU
Dresden 1 (BWR)	206	.099	20.4
Ginna (PWR)	121	.382	46.2
Big Rock Point (BWR)	85	.131	11.1
Point Beach 1 and 2 (PWR)	120	.377	45.3
Oyster Creek 1 (BWR)	183	.186	34.0

TOTALS

Total PWR Assemblies	Total BWR Assemblies	Total MTU
241	474	157.0

Table 3.3

Commercial Fuel Reprocessed at NFS

	MTHM[a]
Rock Point 1	24.3
Yankee Rowe 1	88.8
Indian Point 1	39.4[b]
Humboldt Bay	20.9
Bonus	4.1
Pathfinder	9.6
Dresden 1	71.6
CVNPA	3.5
TOTAL	262.2

[a]Uranium and plutonium.
[b]Includes 16 MT thorium plus uranium (85% U-235 at discharge).

3.1 REFERENCES

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4. PROJECTED COMMERCIAL SPENT FUEL INVENTORY

4.1 Introduction

The single most important factor in developing a projection of commercial spent fuel is the projection of nuclear generating capacity. From a given nuclear generating capacity projection one can generate a projection of fuel discharged by making assumptions on the value of such parameters as the power plant capacity factor, fuel burnup, and net plant efficiency. These parameters are not constant throughout the life of the plant and computer codes have been utilized in making projections which take into account the variability in these parameters. However, to illustrate the approach we may choose typical or steady state values for these parameters and calculate the fuel discharged per year for a typical large (1 GWE) plant. The simplest calculation of this type is the following:

A = metric tons of heavy metal discharged per Gigawatt-electric year,

D = days in the cycle,

C_f = plant capacity or plant load factor,

E_f = net electrical conversion efficiency of the plant,

B = burnup in megawatt-days (thermal)/metric ton of heavy metal,

and $$A = \frac{D \times C_f \times 10^3}{E_f \times B}$$

Typical values for these parameters are[1]

D = 365.25 days
C_f = .7
E_f = .32
B = 30,000 MWD/MTHM for a PWR

which give a value for A of about 27 MTHM/GWE-yr. Hence under the assumptions made about burnup, efficiency, and load factor a typical 1 GWE PWR plant will discharge about 27 metric tons of heavy metal per year. A typical PWR fuel assembly contains about 460 kg of heavy metal so that about 58 assemblies would be discharged per year. A typical BWR would be operated with burnups on the order of 27,500 MWD/MTHM[2] and each assembly contains about 183 kg of heavy metal. Hence any projection of spent fuel derived from a nuclear generating capacity projection must also take into consideration the ratio of PWR's to BWR's that make up the capacity. However, the projection of installed nuclear generating capacity is the starting point or basis for generating spent fuel and nuclear waste projections.

4.2 Discussion of Spent Fuel Projections

A number of spent fuel projections have been identified in the literature. These and the nuclear generating capacity projections used to derive them are discussed.

Figure 4.1 shows several nuclear generating capacity projections that have been used in the various reports reviewed for this inventory report. The time frame is from the year 1975 to year 2000 and these projections were made during the period 1972 to 1980. The general trend has been one of decreasing nuclear generating capacity growth with some growth evident in all but the projection derived[a] from the IRG report[3] low growth case. Nearly all of these projections assume some amount of spent fuel in storage for the beginning year of the projection. Spent fuel is presently stored in the reactor spent fuel storage pools with some storage at the General Electric Morris Illinois facility[4] and the NFS West Valley New York site.[5]

As can be seen from Figure 4.1 there is a wide variation in the projected growth of nuclear power. For purposes of developing an inventory projection of commercial spent fuel it is prudent to select among the more recent projections available which tend to use lower electrical demand growth rates and lower projected nuclear generating capacity. The lower nuclear generating capacity is due to many factors which are beyond the scope of this report. The projection of spent fuel described in Section 4.3 of this Report is based on the DOE Energy Information Administration (EIA) mid-case projection of 1980.

Projections of Radioactive Wastes to be Generated by the U.S. Nuclear Power Industry, ORNL-TM-3965, 1974[7]

This is the earliest report on commercial spent fuel and radioactive waste projections reviewed for this work. The projection of nuclear generating capacity used in this report is derived from plant operating data from 1970 to 1972, the AEC projections of December 1972 [WASH-1139(72)] to the year 2000, and the ORNL ORSAC code beyond year 2000. The ORSAC code allowed for competition between fossil fueled, HTGR, LWR, and LMFBR plants on the basis of economic factors. For the purposes of projecting spent fuel quantities to the year 2000 however, the ORSAC code projection is not relevant.

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[a]The IRG report lists spent fuel generated for two cases. The first, the low growth case, is characterized by 148 GWE installed capacity by the end of 2000 and the second is characterized by 380 GWE installed capacity by 2000. The second case is the ERDA projection of April 1977. The low growth nuclear capacity projection was derived from the IRG spent fuel listing by assuming that 23.65 MTHM is discharged per GWE-yr.

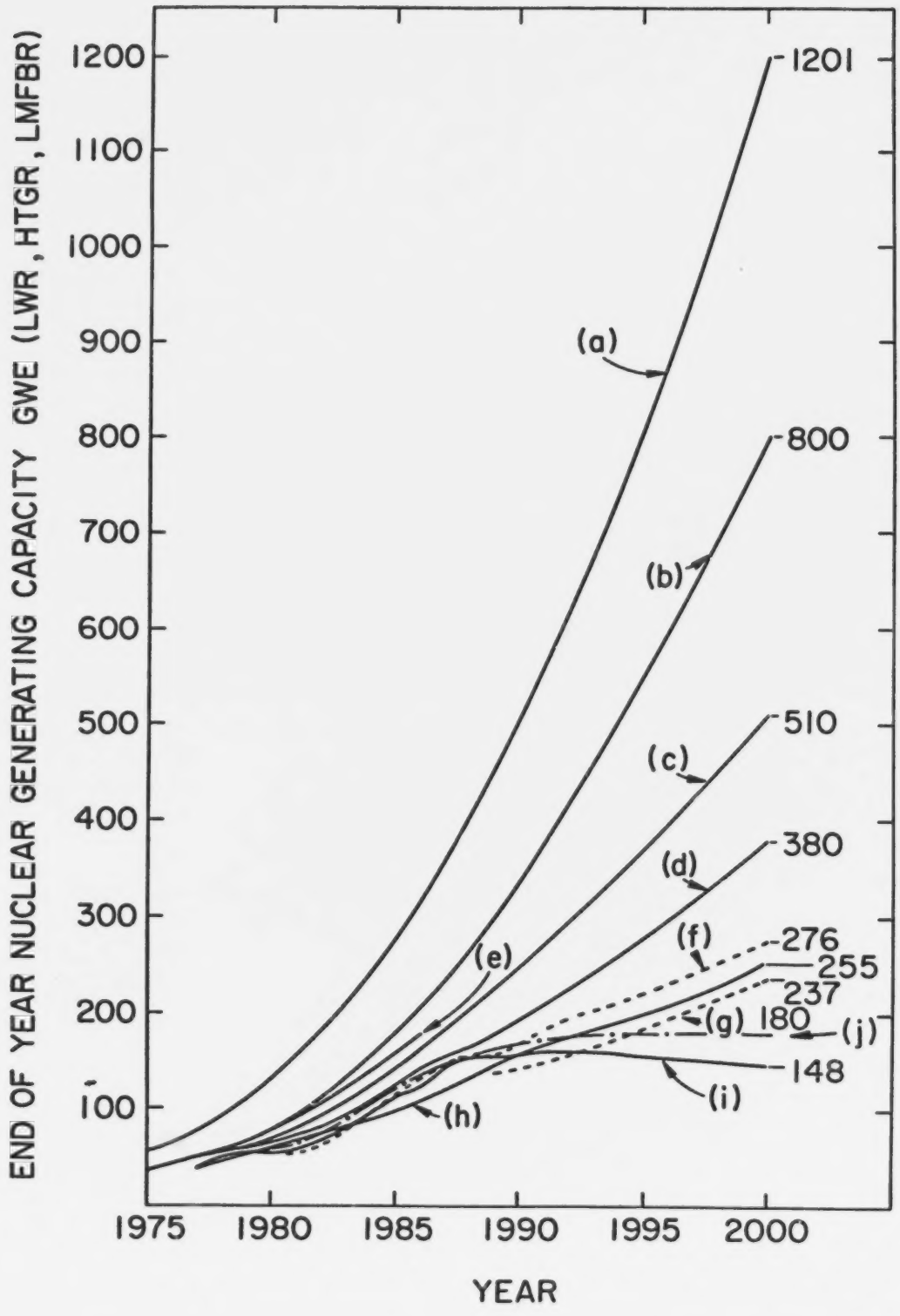

Figure 4.1 Nuclear generating capacity projections.

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Key to Figure 4.1

(a) AEC December 1972 used in ORNL/TM-3965

(b) ERDA December 1975 used in ERDA-76-43

(c) ERDA August 1976 used in ORNL/TM-5472

(d) ERDA April 1977 used in ORNL/TM-6008

(e) Nuclear Assurance Corporation April 1977

(f) DOE/EIA 1979 "High Growth" projection to 1995 extended to year 2000

(g) DOE/EIA 1979 "Low Growth" projection to 1995 extended to year 2000

(h) DOE July 1978 used in DOE/ET-0028

(i) IRG Report "Low Case" October 1978 (derived)

(j) DOE/EIA 1980 Mid Case used in ORO-778

The projection of installed nuclear capacity used in this report is the largest of those in Figure 4.1 and reaches 1201 GWE in year 2000. This projection assumed central station steam power plant (all fuels) capacity growth rates of approximately 8% per year between 1975 and 1980 and decreases to about 5% by year 2000. Nuclear capacity growth was about 20% per year up to 1980 and decreased to about 7.5% at the year 2000. In retrospect these seem extremely high, seriously overestimating nuclear power growth and, of course, not capable of anticipating the energy crisis of 1974 and the events that followed. Other assumptions used in this report were that Pu recycle would begin in 1979, commercialization of the HTGR would begin about 1980 (2.2 GWE), and commercialization of the LMFBR would begin about 1988 (3.5 GWE).

This report was updated in 1976 using a different nuclear generating capacity projection and refined calculations for the waste projections. The updated report ORNL-TM-5427 will be discussed below.

Alternatives for Managing Wastes from Reactors and Post Fission Operations in the LWR Fuel Cycle. U.S. Energy Research and Development Administration, May 1976[8]

The nuclear generating capacity projection used in this report was the ERDA Office of Planning and Analysis projection of 1974 [WASH-1139(74)]. This projection reaches an installed nuclear capacity of 800 GWE at the end of 2000. The total electrical capacity growth is about 6% per year while the nuclear capacity growth begins at 4.5% to 1980, increases rapidly to about 20% per year, and then declines to about 6.5% at year 2000. Of the total nuclear generating capacity of 800 GWE in year 2000, 639 GWE is derived from LWR's with a ratio of PWR's to BWR's = 2:1. Pu recycle is assumed to begin in 1981 and commercialization of the LMFBR is assumed to begin about 1985. Nuclear power plants are all assumed to have a life of 30 years with plant capacity factors that begin at 65%, increase to 70% during years 3 to 15 and then decline thereafter to 40% in the last year.

Since this projection was made there has been a change in electrical energy demand as well as delay and uncertainty concerning the construction of nuclear power plants. For these reasons, the projections of spent fuel and HLW given in this report are not considered further.

Updated Projections of Radioactive Wastes to be Generated by the U.S. Nuclear Power Industry. ORNL/TM-5427, December 1976[9]

The projection of nuclear power growth used in this report is the so-called ERDA "Mid Case" forecast of August 1976 which reaches an installed generating capacity of 510 GWE in year 2000. Of this capacity, 468 GWE is from LWR's with a PWR to BWR ratio of 2:1. Additional assumptions used in this projection are the following:

1. LMFBR's will be commercially introduced in 1995 (2.3 GWE) and reach 6% of the total nuclear market by year 2000.

2. HTGR capacity will increase after 1994 (.3 GWE) and reach 2% of the non-breeder market in 1999 (8.0 GWE).

3. The bulk (92%) of the nuclear capacity in year 2000 will be from LWR's.

4. Fuel reprocessing begins in 1981.

5. Pu recycle begins in 1983.

6. Nuclear capacity factors are:
   1st year, 40%
   2nd and 3rd year, 65%
   4th through 15th years, 70%
   16th through 40th year, declines 2% per year to 40%.

While these assumptions are the most reasonable thus far described, they are not realistic. The nuclear generating capacity projection is high in view of conditions today. Assumption 4, reprocessing beginning in 1981, will clearly not materialize due to executive policy in support of U.S. non-proliferation objectives. Assumption 5, Pu recycle beginning in 1983, is equally unclear and most likely will not remain valid. The schedule of HTGR capacity in Assumption 2 will clearly not be met. The fast breeder reactor projection cannot be met unless a major change in policy and funding occurs in this area.

In spite of the number of invalid assumptions described above, this report gives a comprehensive technical treatment of inventories of nuclear wastes to be generated by the commercial sector under the capacity projection described above. While the emphasis in this work is HLW and spent fuel, this report predicts quantities of eleven types of radioactive wastes. The projections are based on first developing mass flow data for the fuel cycle using the ORNL KWIKPLAN code. The radiation properties of the spent fuel and reprocessed wastes are derived from the ORNL ORIGEN code. Process flow sheets for the various processing steps (reprocessing, fuel fabrication, etc.) are used to estimate the quantities of waste generated in accordance with the mass flows developed by the KWIKPLAN code.

The data of interest presented in the main body of the report are the following:

High level waste from reprocessing LWR, HTGR, and LMFBR fuels

- Radioactivity
- Thermal power
- Volume

Intermediate level TRU waste - waste other than HLW and cladding waste with activity greater than 10 n Ci/gm and surface dose rates from radiation in excess of 10 mrem/hr.

- Volume
- Actinide mass

- Radioactivity
- Thermal power
- Grams, curies, watts of heavy elements as a function of age

Low level TRU waste - waste other than the above having activity greater than 10 n Ci/gm but surface dose less than 10 mrem/hr.

- Volume
- Actinide mass
- Radioactivity
- Thermal power
- Grams, curies, watts of heavy elements as a function of age.

The two appendices contain useful reference data concerning HLW and cladding waste resulting from the reactor and fuel types used in the report. Appendix A has tables of curies of heavy isotopes and fission products in HLW from reprocessing one metric ton of fuel from each of six types of reactor fuels (PWR-U, PWR-Pu, BWR-U, BWR-Pu, HTGR, and LMFBR) as a function of time. A separate table gives the thermal power as a function of time. Appendix B gives the corresponding information for fuel cladding wastes.

Projections of Spent Fuel to be Discharged by the U.S. Nuclear Power Industry, ORNL/TM-6008, November 1977[2]

This report presents the properties of spent fuel to be generated by the commercial power sector to the year 2031. There are actually three projections given; one of these is for the total inventory as generated, and the others are for inventories at a Spent Unreprocessed Fuel Facility (SURFF). Included in the report, for comparison purposes, is a projection of spent fuel to the year 1986 developed by the Nuclear Assurance Corporation in April 1977.

The nuclear generating capacity projection used in this report is the ERDA April 1977 "Low Growth Case" through the year 2000 and a Battelle-Pacific Northwest Laboratories projection thereafter. Of concern here is the projection to 2000 only and some of the assumptions in that projection are:

- Installed nuclear capacity reaches 380 GWE in year 2000.
- All nuclear capacity is from LWR's with a capacity ratio of PWR's to BWR's that is slightly greater than 2 to 1.
- Reprocessing is deferred indefinitely.
- Plant capacity factor is 42% the first year, 65% during the second and and third year, 70% during the 4th through the 15th year, and declines 2% per year to 40% in the final years. This is equivalent to a forty year lifetime capacity factor of 56%.

These assumptions all seem reasonable except that the forecast of nuclear generating capacity is high in comparison to very recent forecasts by DOE in 1979. However, this report contains useful information derived from the ORIGEN2 code on the properties of both PWR and BWR spent fuel assemblies. These tables are based on the reference PWR and BWR assemblies described in ERDA-76-43.[8]

Average fuel burnup was 33,000 MWD/MTHM for PWR's and 27,500 MWD/MTHM for BWR's. The tables give grams of fission products and heavy elements; curies of fission products, light elements and heavy elements; watts; and photon spectrum for both types of spent fuel assemblies.

For purposes of reference and comparison the forecast of nuclear electric generating capacity for the years 1975 to 2000 is shown in Table 4.1. The projection of nuclear installed capacity in Table 4.1 has been used in other work concerning nuclear waste management. The NUS Corporation used this projection in ONWI-20[10] to project quantities of waste other than HLW and spent fuel resulting from reactor and fuel fabrication facilities. In the work done by S. M. Stoller, Division of A. D. Little, Inc. for the Environmental Protection Agency[11] the reference projection is slightly above the ERDA "Low Growth" case until 1993 when it increases more rapidly to reach 450 GWE in year 2000. This increase is due to reduced uncertainty in the nuclear option, increased investor confidence, and selection of nuclear over coal for economic reasons. In addition the authors state that the Stoller projection is reasonable and at the same time provides a degree of conservation for purposes of waste inventory projections.

Included in ORNL/TM-6008 is a projection developed by the Nuclear Assurance Corporation in April 1977.[12] This is shown in Figure 4.1 and reaches a capacity of 176 GWE in 1986. This projection was based on utility practices concerning fuel discharges and expectations for additional commitments for nuclear power plants. In addition, the NAC projection is based on actual reactors whereas most other long range projections use "reference" reactor parameters.

The DOE Energy Information Administration (EIA) High and Low Growth Projections of 1979[13]

These projections are shown in Figure 4.1 with the dashed lines. The high projection reaches an installed nuclear generating capacity of 276 GWE in year 2000 while the low growth scenario reaches 237 GWE at that time. These projections are based on the DOE/EIA projections to year 1995 with the period from 1995 to 2000 based on the DOE/EIA Long Range Energy Assessment Program's Series C Extension which has an annual capacity addition of 10.4 GWE/year for 1995-2000. These projections do not include any consideration of the consequences of the incident at Three Mile Island.

The S. M. Stoller Corporation projection using their DISFUL code is similar to these projections. The DISFUL code projection and the DOE/EIA projections are based in part on operating reactors, those under construction, and future plants announced by utilities. The SMS DISFUL projection reaches 177.7 GWE in 1993 which is about 4 GWE above the DOE low-growth projection for that year.

Technology for Commercial Radioactive Waste Management, DOE/ET-0028, May 1979

This is a fairly comprehensive report that describes the entire high level, spent fuel, and TRU waste management system for the commercial fuel cycle. Two projections were used in this report, the reference projection reached 400 GWE

Table 4.1

Forecast of U.S. Nuclear Electric Power Generating Capacity, GWE, Used in ORNL/TM-6008

Year	Installed Capacity BWR	PWR	TOTAL
1975	15.2	21.8	37.0
1976	17.0	25.7	42.7
1977	17.8	33.4	51.2
1978	18.6	37.5	56.1
1979	20.4	40.2	60.6
1980	22.0	43.8	65.8
1981	23.3	47.5	70.8
1982	25.5	56.8	82.3
1983	28.1	65.4	93.5
1984	32.7	76.5	109.2
1985	37.7	88.9	126.6
1986	42.3	98.3	140.6
1987	46.5	106.7	153.2
1988	50.7	115.1	165.8
1989	55.2	124.1	179.3
1990	60.2	134.2	194.4
1991	65.4	144.8	210.2
1992	70.9	156.1	227.0
1993	76.4	167.4	243.8
1994	82.6	180.1	262.7
1995	89.1	193.4	282.5
1996	95.6	206.7	302.3
1997	102.1	220.0	322.1
1998	108.7	233.2	341.9
1999	115.2	246.5	361.7
2000	121.1	258.6	379.7

in year 2000 while the low-growth projection reached 255 GWE in year 2000 (low growth plotted on Figure 4.1). For the reference projection a number of fuel cycle options were analyzed and these are:

Case 1 Once-through cycle - respository available in 1985.
Case 2a Uranium-only recycle with Pu in the SHLW - repository available in 1985.
Case 2b U-only recycle with Pu stored as PuO_2 - repository available in 1985.
Case 3a U and Pu recycle - repository available in 1985.
Case 3b U and Pu recycle - repository available in 2000.
Case 4a Once-through cycle - repository available in 2000.
Case 4a Defer decision to implement U and Pu recycle to 2000.
Case 4b Defer decision to implement once-through fuel cycle to 2000.

For the low-growth projection only two cases were analyzed and these are:

Case 1 Once-through cycle - repository available in 1985.
Case 3a U and Pu recycle - repository available in 1985.

For the purpose of developing a commercial spent fuel inventory the only case of interest is Case 1. The analysis of the logistics of spent fuel interim storage, packaging, and shipment to the disposal site are also not of immediate interest in developing an inventory of spent fuel.

The two nuclear generating capacity projections to the year 2000 are shown in Table 4.2. The assumptions used in these projections are the following:

Table 4.2

Reference and Low-Growth Projections from DOE/ET-0028

	Reference Projection			Low-Growth Projection		
	Nuclear	Spent Fuel, MTHM		Nuclear	Spent Fuel, MTHM	
Year	Capacity (GWE)[a]	Annual Discharges	Cumulative Discharges	Capacity (GWE)	Annual Discharges	Cumulative Discharges
1977	36.1	700	1,780	36.1	700	1,780
1980	60.1	1,270	6,930	60.1	1,270	6,930
1985	127	2,480	16,300	100	2,060	15,300
1990	194	4,070	38,700	157	3,260	28,800
1995	293	6,210	60,200	200	4,330	48,700
2000	400	8,740	98,600	255	5,700	74,200

[a]End of year capacity.

- All nuclear capacity is from LWR's with 2/3 PWR's and 1/3 BWR's.

- Plants have a 40-year lifetime with capacity factors of 40% initially, increasing to 70% in the fourth year, operating at 70% for 22 years, and then declining linearly to 40% capacity in the fortieth year. The average capacity factor is 62.5%.

- The low-growth projection is based on the DOE/EIA low range estimate of July 1978 which places it essentially between the 1979 DOE/EIA high and low case as shown in Figure 4.1.

- In all cases with reprocessing (Cases 2 and 3 for both projections) it is assumed that reprocessing begins in 1981.

The Interagency Review Group (IRG) Report, October 1978[3]

Appendix D of the IRG report described the scope and magnitude of the waste management system through the use of two cases. These cases were chosen to delineate broad boundaries for the whole waste management system and as such tend to illustrate the extreme points of what may be expected. The higher projection assumed an installed capacity of 380 GWE in year 2000. This is the same as the ERDA April 1977 projection used in ORNL/TM-6008. The low case projection was characterized by having 148 GWE installed nuclear capacity by the year 2000. The IRG low case shown in Figure 4.1 was derived from the IRG Report Table 22 which listed spent fuel generated each year. This projection is given here only to indicate the lowest expected projection of nuclear generating capacity barring a moratorium on nuclear plants.

Spent Fuel Storage Requirements - The Need for Away-From-Reactor Storage DOE/ET-0075, February 1979[14]

This document addresses the need for away-from-reactor storage for spent fuel from the commercial nuclear power sector. The data in the report represents information provided by utilities on their spent fuel storage capabilities and plans. The analysis of storage requirements is based upon a number of factors that affect the quantity of fuel generated and therefore the quantity to be stored at either the reactor sites or at government owned away-from-reactor storage sites. The factors taken into consideration are:

- Spent fuel expansion at reactors.
- Trans-shipment between utilities.
- Reactor capacity factors.
- Discharge capability (full core reserve or annual reload only).
- Reactor fuel burnup.

The sensitivity of the results to the variations in the above factors are presented along with a projection of spent fuel from foreign sources that may be stored in the U.S.

The projection of nuclear generating capacity used in this report is shown in Table 4.3.

Table 4.3

SMSC[a] % of Completion Model for DOE/ET-0075

Year	Nuclear Capacity (GWE)
1983	95.8
1988	167.5
1993	181.6

[a]SMSC - S. M. Stoller Corporation data. SMSC was also involved in the preparation of DOE/ET-0075.

This projection is not plotted in Figure 4.1 although it is close to the DOE/ET-0028 projection as it nears 1993.

Status of Commercial Nuclear High Level Waste Disposal EPRI, September 1976[15]

This report gives an overview of the management of high level nuclear wastes from the commercial nuclear fuel cycle. The projection of nuclear capacity used here is one attributed to ERDA in 1976 and reaches 625 GWE in year 2000. This projection is not plotted in Figure 4.1 and is mentioned here for information purposes only.

Final Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuels, NUREG-0575[16]

This report analyzes the results of various options and mixes of options for providing storage of LWR spent fuel assemblies to the year 2000. Among the alternatives discussed are compacted storage at existing reactor sites, trans-shipment to nearby utility owned sites, trans-shipment to any other reactor site, and utilization of away-from-reactor (AFR) storage facilities.

The report uses the DOE/EIA projection to year 1995 which is extrapolated to 2000 (straight line) to reach an installed capacity of 230 GWE in year 2000 as the reference case. A higher projection, reaching 280 GWE in year 2000 is used only for sensitivity analysis and comparison. These two projections are essentially the two DOE/EIA projections of 237 GWE and 276 GWE shown in Figure 4.1.

The methodology used to predict reactor discharge of spent fuel used in this report is based on operating reactor experience. This approach is therefore in a sense independent of plant capacity factor and fuel burnup assumptions used in "reference" reactor fuel discharge models. The assumptions used in estimating spent fuel discharges are given below.

Number of years in operation following date of commercial operation	Fraction of core discharged per year	
	PWR	BWR
1	0.0	0.0
2-5	0.25	0.20
6	0.33	0.20
7-29	0.33	0.25
30	1.0	1.0

To estimate the spent fuel discharged, the data from operating, under construction, and announced reactors was used. To reach the projected level of nuclear capacity by the year 2000, unplanned "reference" PWR's and BWR's were added in the ratio of 2 PWR's to 1 BWR.

The nuclear generating capacity projection is shown in Table 4.4. The difference between the installed capacity and the capacity discharging is due to the length of time between fuel loading and the first discharge. A one-year period between the initial fuel loading and the start of operation is assumed and hence the first discharge occurs three years following fuel loading. The fuel discharge schedule resulting from these assumptions is shown in Table 4.5.

The DOE/EIA Mid-Case Projection of July 1980

The Energy Information Administration's mid-case projection is shown in Figure 4.1 as reaching 180 GWE in year 2000. This projection was given in the EIA Annual Report to Congress for 1979 dated July 1980. This nuclear growth projection is shown in Table 4.6.[17] This projection was also used by Carter et al. in their report (ORO-778) on nuclear waste inventories.[18]

This projection is based on an increase in electrical demand but at a lower rate than past projections. The average growth rate for demand is projected to be about 2.6% yearly up to the year 2000. This is a decrease over growth rates of 3.2% and higher for the years prior to 1978. This decrease reflects the lower growth in GNP, the increase in conservation, and the growth of dispersed generation for which base loaded large nuclear plants are uncompetitive. Additional uncertainties, peculiar to nuclear plants such as site suitability and the accident at Three Mile Island are projected to reduce the demand for nuclear generated electricity. However, the recent lower projections of nuclear capacity and the reasons suggested for them may not be valid over the long term.

4.3 Projected Spent Fuel Inventory

The DOE/EIA projection of July 1980 has been selected as the basis for the projected spent fuel inventory for this report. This nuclear capacity projection reaches 180 GWE in the year 2000. The spent fuel inventory resulting from this generating capacity projection is reported in ORO-778.[18] This projected spent fuel inventory is based on a DOE report on the need for away-from-reactor storage capacity (DOE/NE-0002, Reference 19). DOE/NE-0002 assumes that reactors

operate at 70% capacity factor with an average fuel burnup of 23,000 MWD/MTU for BWR's and 31,000 MWD/MTU for PWR's. Reactors are assumed to have a 40-year life. The ratio of PWR's to BWR's is approximately 2 to 1. Table 4.7, taken from ORO-778 shows the projected spent fuel inventory resulting from these assumptions. The cumulative spent fuel in storage at the end of 1979 is given in Table 4.7 as 5930 MTU which is higher than the existing spent fuel in storage inventory of about 5190 MTU given in Section 3 above. This is because the existing spent fuel inventory of Section 3 is based on actual reported values to the end of 1979 whereas the data supplied in DOE/NE-0002 (the source of data for Table 4.7) appears to have actual values up to about May 1979 and then projected spent fuel discharges to the end of 1979.[20]

Table 4.4

Nuclear Generating Capacity Installed and Discharging Spent Fuel for Each Year, 1979-2000 From NUREG-0575

Year	Capacity Installed, GWE	Capacity Discharging, GWE
1979	58	46
1980	66	48
1981	73	51
1982	80	58
1983	87	66
1984	94	73
1985	102	80
1986	110	87
1987	119	94
1988	125	102
1989	134	110
1990	142	119
1991	151	125
1992	160	134
1993	168	142
1994	177	151
1995	187	160
1996	195	168
1997	202	177
1998	212	187
1999	221	195
2000	230	202

Table 4.5

Annual and Cumulated Schedules of Spent Fuel Discharge
From NUREG-0575

Year	GWE Capacity Discharging	Annual Discharge, MTHM	Cumulative Discharge,[a] MTHM
1979	40	1,420	---
1980	48	1,520	2,900
1981	51	1,640	4,600
1982	58	1,850	6,400
1983	66	2,100	8,500
1984	73	2,300	11,000
1985	80	2,440	13,000
1986	87	2,650	16,000
1987	94	2,840	19,000
1988	102	3,050	22,000
1989	110	3,300	25,000
1990	119	3,600	29,000
1991	125	3,720	32,000
1992	134	3,950	36,000
1993	142	4,200	41,000
1994	151	4,380	45,000
1995	160	4,620	50,000
1996	168	4,840	54,000
1997	177	5,100	60,000
1998	187	5,460	65,000
1999	194	5,730	71,000
2000	202	5,800	77,000

[a]Does not include about 4700 MT of spent fuel discharged prior to 1979 and stored AR and AFR at the end of 1978.

Table 4.6

DOE/EIA Mid Case Projection (July 1980)

Year	(GWe)
1979	51.9
1980	56.5
1981	61.4
1982	75.7
1983	91.5
1984	104.1
1985	125.0
1986	136.0
1987	148.5
1988	154.8
1989	159.9
1990	167.4
1991	172.4
1992	174.8
1993	177.2
1994	178.5
1995	178.5
1996	178.5
1997	178.5
1998	178.5
1999	178.5
2000	180.0

Table 4.7

Yearly Discharges and Spent Fuel Accumulation, MTU (From ORO-778)

Calendar year	BWR		PWR		Total	
	Annual	Cumulative	Annual	Cumulative	Annual	Cumulative
1979		2.545		3,385		5,930
1980	625	3.170	905	4,290	1,530	7,460
1981	625	3,795	1,020	5,310	1,645	9,105
1982	730	4,525	1,070	6,380	1,800	10,905
1983	910	5,435	1,290	7,670	2,200	13,105
1984	1,020	6,455	1,620	9,290	2,640	15,745
1985	1,175	7,630	1,810	11,100	2,985	18,730
1990	1,820	16,010	2,710	23,360	4,530	39,370
1995	1,925	25,565	3,095	38,335	5,020	63,900
2000	1,925	35,190	3,095	53,810	5,020	89,000

To obtain the characteristics of spent fuel any number of references may be consulted. ORNL/TM-6008 reports results for a plant capacity factor of 80% and burnups of 27,500 MWD/MTU for BWR's and 33,000 MWD/MTU for PWR's. These assumptions are sufficiently close to the above assumptions on plant capacity factor and burnups to allow us to use the ORNL/TM-6008 tables as an approximate reference for the characteristics of spent fuel. The appropriate tables from ORNL/TM-6008 are reproduced in Appendix E to provide a handy reference.

4.4 References

1. Technology for Commercial Radioactive Waste Management, U.S. Department of Energy, DOE-ET-0028, May 1979.

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5. EXISTING AND PROJECTED TRU INVENTORY

5.1 Introduction

Transuranic contaminated wastes, or TRU wastes, are materials that are contaminated with transuranic nuclides having relatively long half lives. As currently classified, TRU wastes are waste materials containing greater than 10 nanocuries of certain alpha emitting long lived radionuclides per gram of waste. The TRU radionuclides are defined as U-233 and all transuranium nuclides except Pu-238 and Pu-241; however, in practice these last two isotopes are often part of TRU waste. Most TRU wastes have sufficiently low gamma and beta activity that they may be handled without any special precautions. Such wastes are classified as "contact handled" TRU wastes and these wastes have a surface dose rate less than 200 mR/hr. Other TRU wastes have sufficiently energetic gamma and beta activity to require remote handling. These wastes are classified as "remote handled" TRU wastes and have a surface dose rate greater than 200 mR/hr. Most TRU wastes have been generated in defense related activities of the DOE at Hanford, Savannah River, Los Alamos Scientific Laboratory (LASL), and the Rocky Flats plant.[10] Smaller quantities are generated at Argonne National Laboratory (ANL), Oak Ridge National Laboratory (ORNL), Mound Laboratories, Bettis Atomic Power Laboratory (BAPL), Lawrence Livermore Laboratory (LLL), Battelle Columbus Laboratory (BCL), and the Idaho National Engineering Laboratory (INEL). The commercial nuclear power industry produces little TRU waste since there is no reprocessing and Pu recycle permitted under current policy. A small amount of TRU waste is also generated by industrial and research activities. Prior to 1970 it was common practice to bury TRU wastes along with Low Level Wastes (LLW) at both DOE and commercial burial sites. In 1970 an Atomic Energy Commission (AEC) policy directive mandated that TRU wastes be placed in retrievable storage. No TRU wastes have been buried at commercial burial sites since late 1970. While this section is primarily concerned with DOE TRU wastes, the existing inventory of commercial TRU wastes is given at the end of this section.

5.2 Description of TRU Wastes

TRU wastes exist in a variety of physical forms. The DOE defense programs generate metal scrap, paper and rags, sludges, and filters. The sludges are generated in various processes associated with the fabrication of nuclear weapons. Other programs such as the production of plutonium and curium heat sources, the thorium fuel cycle development program, and minor isotopes production also generate TRU wastes.[2] The bulk of the waste at commercial burial grounds is in the form of paper trash, filters, broken glassware, rags, cleaning aids, defective equipment and similar materials whose surfaces have been in contact with TRU nuclides. Table 5.1, updated[3] from a PNL report[4] on TRU waste characteristics, shows the contents of some important TRU wastes stored at Idaho from 1971 to 1979. The remaining volume of TRU waste stored at Idaho is in about 86 different waste forms. Among these are retrieved drums, concrete, asphalt, sand, and glass waste. Idaho has the largest volume of retrievably stored waste in the DOE system. Very little TRU waste is generated at Idaho, but TRU wastes are shipped to Idaho from Rocky Flats, Mound Laboratories, ANL, BAPL, and BCL.

Table 5.1

Volumes and Pu Contents for Important Waste Forms Stored at Idaho From September 1971 Through December 1979

Waste Form	Volume		Pu Content
	m^3	% of Total	% of Total
Metal Scrap	9264	30	13
Paper and Rags - Dry	3474	11	3
Second Stage Sludge	1635	5	1
Paper and Rags - Moist	1556	5	1
Filters - CWS	1881	6	2
First Stage Sludge	1364	4	13
Raschig Rings - Unleached	328	1	5
Incinerator Ash - Leached	21	0.1	8

A partial list of TRU waste forms, therefore, includes metal scrap, sludges, paper, rags, filters, decontamination and decommissioning (D&D) materials (lumber, concrete, etc.), process equipment (tanks, pumps, piping, etc.), clothing, tools, incinerator ash, glassware, biological wastes, resins, instrumentation, and plastic and rubber components.

Very little is available in the literature on the chemical composition of most defense TRU waste forms. Work done by LASL, Rocky Flats (RFP), and Pacific Northwest Laboratory (PNL) characterized the process sludges from research and production activities at RFP and LASL and reported these in Reference 4. The sludge resulting from these processes contains about 50 to 60% water, a variety of metallic compounds, and diatomaceous earth filter materials. The new RFP production facility, not yet operational, is expected to generate a sludge containing only about 10 to 15% water. Table 5.2, from Reference 4, shows the composition of the existing RFP and LASL process sludges.

Table 5.2

RFP and LASL Sludge Analysis

Analysis	Average (ppm)	
	RFP	LASL
Al	9,200	27,500
Ca	130,500	165,000
Fe	46,500	55,000
K	6,200	11,000
Mg	12,700	16,500
Na	70,100	16,500
Si	75,000	187,000

Table 5.2 Continued

NO_3	67,700	--
CO_3	11,200	--
Cl	6,000	--
H_2O	505,000	--
Pu	22	--
Am	13	--
U	820	--
P	--	22,000
density	1.0 gm/cm3	--

Since incineration of contaminated combustible waste is being developed and practiced at several DOE facilities, some work has been done on characterizing incinerator ash. A representative incinerator feed composition, extracted from Reference 4, is given in Table 5.3.

Table 5.3

Typical Incinerator Feed Material

Material	Composition Weight %
Paper	15
Rags	15
Wood	10
Neoprene	10
Hypalon	10
Latex	5
PVC	20
Polyethylene	15

The ash resulting from the incineration of combustible wastes is primarily dependent on the feed mixture and to a lesser extent upon the incinerator design and operating conditions. The ash composition from several incinerators now under development is reported in Reference 4, and is reproduced here in Table 5.4.

The transuranic content of retrievably stored TRU wastes at Savannah River (as of 10/1/77) are reported in Reference 2 and reproduced here in the right hand column of Table 5.5. The data for Idaho (as of 12/31/79) are from Reference 3.

Table 5.4

Residue Compositions From Selected Incinerators (wt%)

	RFP/FBI	Mound Cyclone Incinerator	SRL	LASL CAI	PNL Pyrolysis Incinerator
Al_2O_3	35	33	15	28	17
C	5	--	3	6	14
CaCl	--	--	--	2	4
CaO	--	6	24	2	6
Cr_2O_3	10	3	--	--	--
Fe_2O_3	1	5	11	1	--
MgO	--	8	6	--	--
NaCl	5	--	--	--	2
Na_2CO_3	30	--	--	--	--
SiO_2	10	31	16	35	34
TiO_2	--	6	18	23	17
ZnO	--	3	6	--	--
Total	96	95	99	97	94

Table 5.5

Transuranic Content of Idaho and Savannah River TRU Waste in Retrievable Storage (grams)

Isotope	Idaho	Savannah River
Am-241	18,000	
Cm-244	14	330
Pu-238	2,400	15,000
Pu-239	249,200	32,000
Pu-240	15,600	
Pu-241	1,000	
U-233	68,000	

About 90% of the Pu-238 at SRP originally came from either Mound Laboratories or LASL. Over 10,000 grams of the Am-241 at Idaho is from the first stage sludge generated by Rocky Flats. Additional details on the TRU content of retrievably stored defense wastes are contained in Reference 2.

Commercial burial grounds have never segregated TRU contaminated wastes from other wastes, and the TRU wastes are therefore mixed with large amounts of

low level waste. Also, no TRU waste has been emplaced at commercial burial grounds since 1970. Burial site records do not clearly differentiate between TRU and LLW and so both the inventory data and the chemical composition are difficult to obtain. TRU wastes resulting from the reprocessing of commercial spent fuel at the Nuclear Fuel Services (NFS) West Valley facility include fuel cladding hulls, salt solutions, degraded solvents, wet particulates, precipitates and slurries, dry particulates, and organic liquids.[5]

5.3 Inventories of DOE TRU Waste

5.3.1 Buried TRU Waste

It has been difficult to obtain accurate data on the quantities of TRU contaminated wastes that have been buried at DOE sites. This is due to a number of factors; among these is the fact that accurate and complete records were not always maintained prior to 1970. In addition, TRU wastes were often not segregated from LLW prior to burial so that volume estimates may include some amount of LLW. The volumes of TRU waste that appear in the literature are generally based on the container volume and not on the actual volume of waste.[6] The problem is further complicated by the fact that some of the waste containers have suffered damage over the years and this has led to contamination of the surrounding soil. The volume of contaminated soil is often estimated and included in the inventory estimates where soil contamination was known to have occurred. Consequently, the inventory data on buried TRU waste are generally based on estimates with considerable uncertainty. The above situation also applies to buried commercial TRU waste with the exception that no TRU waste was buried after 1970 in commercial burial grounds.

Several reports from the literature[1,2,10,11,12] and the SWIMS data have been analyzed and compared. The most recent and reliable data are contained in the SWIMS system and the Oak Ridge report ORO-778.[10] The data from these reports on buried TRU waste have been collected and are shown in Table 5.6. The ORO-778 data reports the volume of waste and the volume of contaminated subsurface (soil) separately. These volumes of contaminated soil are based on conservative estimates. Some buried waste has been exhumed as part of experimental waste-retrieval projects. The estimated activity is given for the waste at the time of burial. Reference 1, an early draft of ORO-778 included estimates of the uncertainty of the waste volume. These are no longer reported since they appear to have been included to account for the volumes of contaminated soil that were not separately reported. Nevertheless, due to the reasons discussed above, the uncertainties in some cases are believed to be large. The discrepancies between the SWIMS and ORO-778 data may be due to the difference in the date of the inventory as well as differences in the data sources and methods of estimation.

5.3.2 Stored TRU Waste

The quality of the data for the quantities of TRU waste retrievably stored is considerably better than that for the buried waste. Table 5.7 gives the data on retrievably stored TRU waste. The uncertainty in the volumes of retrievably stored TRU waste is believed to be no more than ±10%.

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Table 5.6

Inventories of TRU Waste Buried at DOE Sites[a]

	ORO-778 (10/1/79)			DOE SWIMS (3/31/80)				
	Waste (m^3)	Contaminated Subsurface (m^3)	Mass of TRU Elements (kg)	Waste Vol. (m^3)[b] CH[c]	RH	Mass of TRU Elements (kg)[d]	Curies (RH Only)	Estimated Activity At Time of Burial (Ci)
Idaho	57,113 (note e)	212,859	381	62,100		375.5		252,975 (CH, 1954-1970, Ref. 3). 583,000 (RH mixed with TRU and LLW, 1954-1964, Ref. 3 and 9).
Hanford (Total)	211,251	291,103	371		314,300	381.8	941,760	
200 Areas					155,200	374.7	931,310	478,150 (RH, Ref. 7)
300 Area					159,100	7.1	10,450	<2,000 (RH, Ref. 7).
Savannah River	27,270	61,000	7		2,638		620	7,400 (1955-1965, Ref. 8).
Los Alamos	11,485	28,300	13	11,420	65.76	5.0	4,349	
Oak Ridge	6,229	167,072	13	37.9	247.7	.13	1,040	
Nevada Test Site	5,700	5,700	.23	36.48	11.0		.004	
Pantex	33.4	33.4		33.44		2.4		
Sandia	3	3		2.826				
TOTALS	319,084	766,070	785.2	73,631	317,300	764.8	947,770	

[a]Inventory as of date given except for ORO-778 which did not specify date but 10/1/79 is assumed here.
[b]SWIMS reports waste buried even though some buried waste was retrieved and then placed in retrievable storage. This is believed to have occurred only at Idaho.
[c]CH = contact handled, surface dose <200 mR/hr.
RH = remote handled, surface dose >200 mR/hr, also called beta-gamma TRU.
[d]Contact handled plus remote handled.
[e]Amount remaining in the ground after retrieval of certain buried wastes.

Table 5.7

Inventories of TRU Waste Retrievably Stored at DOE Sites[a]

	ORO-778 (10/1/79) Waste Volume (m^3)		DOE SWIMS (3/31/80)				Estimated Activity At Time of Burial (Ci)
			Waste Volume (m^3)		Mass of TRU Elements	Curies	
	CH[b]	RH	CH	RH	(kg)[c]	(RH only)	
Idaho	39,571	19	40,540	16.79	362.94	244.1	238,700 (1970-1979, Ref. 3 and 9).
Hanford	8,038	0.8	8,277	20.35	178.59	128,812	
Asphalt Pad			3,642	0.8438	154.2	12.2	1.1 (RH, Ref. 7).
Trench			4,634		19.2		
TRU Caisson			0.927	19.51	5.19	128,800	3,400 (RH, Ref.7).
Savannah River	2,365		2,527		56.68		288,000 (1965-1977, Ref. 8).
Los Alamos	3,528	73.5	3,908	7.743	73.12	160.0	
Oak Ridge	384	789	365	557.3	12.07	15,300.0	
Nevada Test Site	243		206.8		1.52		
Sandia			0.198				
Paducah C-746			2.633		.277		
Argonne National Lab.				2.337	.055	26,980	
TOTALS	54,129	882.3	55,826	604.5	685.25	171,496	

[a]Inventory as of date given
[b]CH - contact handled, surface dose <200 mR/hr.
RH - remote handled, surface dose >200 mR/hr, also called beta-gamma TRU.
[c]Contact handled plus remote handled.

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5.3.3 Additional TRU Waste Data for the Idaho National Engineering Laboratory (INEL) Site

The Idaho National Engineering Laboratory (INEL) is the site of the Radioactive Waste Management Complex (RWMC). This facility is the largest TRU storage area in the DOE system. The activities of the RWMC are described in References 6 and 9. The RWMC is essentially divided into the Transuranic Storage Area (TSA) for the disposal of TRU wastes (activity >10 nCi/gm) and the Subsurface Disposal Area (SDA). At the TSA the wastes are stored in drums and boxes on an asphalt pad. The wastes are then covered with plywood, a plastic liner, and finally with soil. Within the TSA there are storage vaults for retrievably storing remote handled TRU wastes. The SDA was used for the burial of TRU wastes prior to 1970. One area within the SDA is still used for low level beta, gamma wastes. One portion of the SDA, called the Transuranic Disposal Area (TDA), was used for the disposal of TRU wastes with activities <10 nCi/gm. The TDA is now closed. Table 5.8 is an updated version of Table 4.2 of Reference 9. These data were supplied by H. Batchelder of EG&G, INEL. Table 5.9 is reproduced from Table 4.3 of Reference 9.

A substantial effort to reconcile data concerning buried TRU waste at INEL has been conducted. This work involved the collection and review of early waste disposal records. In addition the early waste retrieval program exhumed buried wastes from the SDA. Based on these results the uncertainty in the volume of buried waste is believed to be on the order of ±30% (Reference 3). The uncertainty in the stored waste volume is primarily due to a small amount of nonstandard drums. The uncertainty is believed to be about ± 5% but certainly no more than ± 10%. TRU process wastes from the Idaho Chemical Processing Plant are mixed with the HLW stream and are subsequently calcined.

5.4 Projections of TRU Wastes at DOE Sites

A few projections of TRU wastes have appeared in the literature such as in References 8 and 11. These projections are usually for five or ten years and generally report volumes. ORO-778 (Reference 10) provides TRU waste projections for each DOE facility for the years 1979-2000. These projection data were developed in personal communications between ORNL personnel and personnel at the various DOE sites. These estimates are current as of October 1, 1979. The ORO-778 projections are reproduced here in Tables 5.10 to 5.15. Notes accompanying the tables were extracted from the text of ORO-778.

5.5 Inventories of Commercial TRU Waste

The commercially generated TRU waste includes the waste generated in the nuclear fuel cycle as well as numerous research and development activities and small commercial use of certain TRU radioisotopes. No TRU waste has been emplaced at commercial burial grounds since 1970. The most frequently quoted data on the TRU contaminated waste at commercial burial grounds are those from the TERA Advanced Services Corporation report for Lawrence Livermore Laboratory, Reference 12. ORO-778, Reference 10, uses some data from the TERA report and

Table 5.8

Radionuclide Content of Buried and Stored TRU Waste at INEL

Uranium and Transuranic Radionuclides	Half-Life[a] (Years)	Buried Waste[b,e] Emplaced Curies	Emplaced Percent	As of January 1, 1991 Curies	As of January 1, 1991 Percent	Stored Waste Emplaced (through December 31, 1979) Curies	Emplaced (through December 31, 1979) Percent	As of January 1, 1991 Curies	As of January 1, 1991 Percent
Pu-238	86.4	530.1	0.23	436	0.37	41,749	17.49	148,300	34.98
Pu-239	24,390	19,725	8.37	19,712	16.81	15,306	6.41	30,180	7.12
Pu-240	6,580	4,798	2.04	4,783	4.08	3,729	1.56	7,340	1.73
Pu-241	13.2	167,770	71.25	47,634	40.63	117,000	49.02	124,600	29.39
Pu-242	379,000	0.2	<0.01	0.2	<0.01	0.3	<0.01	0.6	<0.01
Am-241	458	42,586	18.08	44,615[c]	38.05	58,425	24.48	111,209[c]	26.24
Cm-244	17.6	0	0	0	0	1,113	4.66	1,113	0.26
U-233	162,000	0.5	<0.01	0.5	<0.01	643	0.27	1,170	0.27
U-235	710,000,000	0.3	<0.01	0.3	<0.01	0.005	<0.01	0.02	<0.01
U-238	4,510,000,000	68	0.03	68	0.05	0.004	<0.01	0.01	<0.01
Total Radioactivity		235,478	100	117,249	100	238,700	100	423,900	100
Total Weight of TRU (including U-233), g[d]		381,000		355,200		366,000		566,150	
Total Weight of U-235, U-238, kg		203,322		203,322		14.3		24.0	
Total Waste Volume, ft^3		2,190,000		2,018,000		1,417,000		2,390,000	
Transuranic Ci/ft^3		0.115		0.058		0.17		0.18	

[a]Values are taken from the Radiological Health Handbook (Bureau of Radiological Health 1970).
[b]Does not include the beta-gamma waste present. See Table 5.9 for beta-gamma content.
[c]Includes americium-241 produced by decay of plutonium-241 after emplacement.
[d]Because of rounding errors, values shown may differ slightly from values obtained by converting curies of individual radionuclides to grams and adding.
[e]Buried data through 1990 are corrected for retrieval.

NOTE: These data are an update provided by H. Batchelder, EG&G, 9/23/80 of Table 4.2 of DOE/ET-0081 (Revised).

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Table 5.9

Estimated Activity in Beta-Gamma Waste Mixed With Buried TRU Waste at INEL

Beta-Gamma Radionuclide	Half-Life[a] (Years)	Emplaced Curies	Emplaced Percent	As of January 1, 1985[b] Curies	As of January 1, 1985[b] Percent
Co-60	5.27	99,000	17.0	4214.3	56.8
Ni-59	80,000	1,500	0.3	1500.0	20.2
Sr-90	29.0	1,000	0.2	563.5	7.6
Cs-137	30.17	1,000	0.2	576.2	7.8
Mixed activation products (Co-60)[c]	5.27	6,000	1.0	255.4	3.4
Mixed fission products (Cs-137)[c]	30.17	500	0.1	288.1	3.9
Unidentified beta-gamma[d]	3.0	5,500	0.9	21.5	0.3
Short-lived beta-gamma	<<1	468,500	80.3	negligible	0.0
TOTALS		583,000	100.0	7419	100.0

[a]Values are taken from the Chart of the Nuclides, Naval Reactors - U. S. Department of Energy, April 1977.

[b]Although beta-gamma waste was buried with the TRU waste from 1954 through 1964, about half of the activity was buried in 1962 and 1963. Therefore, these simplified calculations of radioactive decay were based on an assumed burial date of January 1, 1961 (24 years of decay by the year 1985) for all of the beta-gamma waste.

[c]Decay calculations are based on the half-life of the dominant radionuclide.

[d]Decay calculations are based on an estimated value of the effective half-life.

<< = Much less than.

Table 5.10

Projected Storage of Retrievable Transuranic Defense Waste at INEL[a]

Fiscal Year	Contact Handled		Remotely Handled		Total Accumulated Volume (m^3)
	Projected Storage Rate (m^3/year)	Accumulated Volume (m^3)	Projected Storage Rate (m^3/year)	Accumulated Volume (m^3)	
1979	---	39,571	---	19	39,590
1980	2,585	42,156	12.65	31.65	42,188
1981	2,658	44,814	12.05	43.70	44,858
1982	2,725	47,539	12.05	55.75	47,595
1983	2,882	50,421	11.05	66.80	50,488
1984	3,035	53,456	11.15	77.95	53,534
1985	2,777	56,233	11.15	89.10	56,322
1986-1990	2,152	66,993	10.15	[illegible]	67,133
1991-1995	1,914	76,563	10.15	[illegible]	76,754
1996-2000	1,914	86,133	10.1'		86,374

[a]Data from Reference 10. Accuracy is believed to be within +20% to -40%. Most wastes are from other sites as described in Section 5.2 above.

Table 5.11

Projected Storage of Retrievable Transuranic Defense Waste at Hanford[a]

Fiscal Year	Contact Handled		Remotely Handled		Total Accumulated Volume (m^3)
	Projected Storage Rate (m^3/year)	Accumulated Volume (m^3)	Projected Storage Rate (m^3/year)	Accumulated Volume (m^3)	
1979	---	8,038	---	0.8	8,039
1980	696	8,734	739	740	9,474
1981	780	9,514	855	1,595	11,109
1982	289	9,803	855	2,450	12,253
1983	1,310	11,113	657	3,107	14,220
1984	405	11,518	653	3,760	15,278
1985	405	11,923	738	4,498	16,421
1986-1990	405	13,948	738	8,188	22,136
1991-1995	405	15,973	738	11,878	27,851
1996-2000	405	17,998	738	15,568	33,566

[a]Data from Reference 10. Data do not include receipts of TRU waste from off-site. The uncertainty in these estimates is believed to be ±25%.

Table 5.12

Projected Storage of Retrievable Transuranic Defense Waste at SRP[a]

Fiscal Year	Contact Handled[b]	
	Projected Storage Rate (m^3/year)	Accumulated Volume (m^3)
1979	---	2,365
1980	146	2,511
1981-1985	146	3,095
1986-1990	146	3,825
1991-1995	146	4,555
1996-2000	146	5,285

[a]Data from Reference 10.
[b]The projected TRU waste at SRP does not include any remotely handled material. The estimated uncertainty in these projections is ±50%.

Table 5.13

Projected Storage of Retrievable Transuranic Defense Waste at LASL[a]

Fiscal Year	Contact Handled		Remotely Handled		Total Accumulated Volume (m^3)
	Projected Storage Rate (m^3/year)	Accumulated Volume (m^3)	Projected Storage Rate (m^3/year)	Accumulated Volume (m^3)	
1979	---	3,528	---	74	3,602
1980	1,418	4,946	7	81	5,027
1981	568	5,514	7	88	5,602
1982-1985	568	7,786	7	116	7,902
1986-1990	568	10,626	7	151	10,777
1991-1995	568	13,466	7	186	13,652
1996-2000	568	16,306	7	221	16,527

[a]Data from Reference 10. These projections include wastes from Sandia, Lovelace Bioenvironmental Research Institute, and LASL. The uncertainty in these estimates is believed to be +150% to -25%.

Table 5.14

Projected Storage of Retrievable Transuranic Defense Waste at ORNL[a]

Fiscal Year	Contact Handled		Remotely Handled		Total Accumulated Volume (m^3)
	Projected Storage Rate (m^3/year)	Accumulated Volume (m^3)	Projected Storage Rate (m^3/year)	Accumulated Volume (m^3)	
1979	---	384	---	789	1,173
1980	35	419	42.5	831	1,250
1981	71.8	491	70.8	902	1,393
1982-1985	71.8	778	70.8	1,185	1,963
1986-1990	71.8	1,137	70.8	1,539	2,676
1991-1995	71.8	1,496	70.8	1,893	3,389
1996-2000	71.8	1,855	70.8	2,247	4,102

[a]Data from Reference 10. The uncertainty in these projections is ±25%. These projections include some small amount (<1 m^3 per year) of TRU wastes from the Paducah Gaseous Diffusion Plant.

Table 5.15

Projected Storage of Retrievable Transuranic Defense Waste at NTS[a]

Fiscal Year	Contact Handled[b]	
	Projected Storage Rate (m^3/year)	Accumulated Volume (m^3)
1979	---	243
1980	31.8	275
1981-1985	31.8	434
1986-1990	31.8	593
1991-1995	31.8	752
1996-2000	31.8	911

[a]Data from Reference 10.
[b]The projected TRU waste at NTS does not include any remotely handled material. Approximately 28% of these wastes will come from Lawrence Livermore Laboratory (LLL). The uncertainty in these volume projections is ±25%.

some from a LASL report (Reference 13). An NRC listing of these quantities shows similar data (Reference 14). All three references are fairly consistent in the volume and kilograms of TRU estimates. Reference 10 does not report Special Nuclear Material (SNM) quantities. The data given in Table 5.16 are from Reference 14.

Table 5.16

Cumulative Waste Inventory at Commerical Burial Sites (1978)

Site	Volume Buried (m^3)	SNM (kg)	TRU (kg)
Beatty, NV	69,943	199.13	14.3
Maxey Flats, KY	135,072	431.8	63.76
Hanford, WA	24,635	113.53	35.55
Sheffield, IL	88,632	56.4	13.4
West Valley, NY	66,826	56.0	4.0

The volumes of waste given in Table 5.16 above are actually the volumes of low level waste that might contain TRU wastes. These wastes are from a variety of waste generators including government contractors, commercial nuclear power plants and facilities, and industrial and research facilities. There are no uncertainty estimates available for the volume estimates of Table 5.16.

ORO-778 provides a projection of industrial TRU wastes.[10] Through 1982 most of these wastes will be generated by facilities involved in fuel fabrication on government projects. The waste projections given beyond 1983 are expected to be entirely from operations in the private sector. The projections of TRU industrial wastes given in Table 5.17 are taken from ORO-778 which are based on Reference 15.

Table 5.17

Projected Generation of Industrial Transuranic Wastes[a]

Year	Wastes Generated In: Government Work (m^3/year)	Wastes Generated In: Continuing Operations (m^3/year)	Accumulated Volume (m^3)
1980	1,092	405	1,497
1981	1,165	493	3,155
1982	1,270	249	4,674
1983	6.2	80.9	4,761
1984-1985	---	76.4	4,914
1986-1990	---	76.4	5,296
1991-1995	---	76.4	5,678
1996-2000	---	76.4	6,060

[a]Data from Reference 10, projections as of January 1980.

5.6 References

1. Spent Fuel and Waste Inventories and Projections. W. L. Carter, B. C. Finney, C. W. Alexander, J. M. McNair and J. O. Blomeke, ORNL/TM-7320, July 1980.

2. Preliminary Evaluation of the Characteristics of Defense Transuranic Wastes. H. C. Shefelbine, SAND 78-1850, November 1978.

3. Personal communication, H. Batchelder, INEL to J. P. Indusi, BNL, September 23, 1980.

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8. Alternatives for Long Term Management of Defense Transuranic Waste at the Savannah River Plant, U. S. Department of Energy, DOE/SR-WM-79-1, July 1979.

9. Environmental and Other Evaluations of Alternatives for Long Term Management of Stored INEL Transuranic Waste, Idaho National Engineering Laboratory, DOE/ET-0081 (Revised), December 1979.

10. Spent Fuel and Waste Inventories and Projections, U. S. Department of Energy, Oak Ridge Operations Office, ORO-778, August 1980.

11. Draft Environmental Impact Statement Waste Isolation Pilot Plant, U. S. Department of Energy, DOE/EIS-0026-D, April 1979.

12. TERA Advanced Services Corporation, Inventory and Sources of Transuranic Solid Waste, UCRL-13934, August 1978.

13. Studies of the Transuranic Waste Storage Under Conditions Expected in the Waste Isolation Pilot Plant (WIPP), October 1, 1977-June 15, 1979, S. T. Kosiewicz, et al., LA-7931-PR, January 1980.

14. Letter to Mr. Alan Grey, EG&G Idaho, Inc. from Sally Latlief, Division of Waste Management, NRC, dated June 22, 1979.

15. Acceptance of Commercial Transuranic Waste, A. L. Tuboas, et al., DOE AL/TRU-8001, February 1980.

APPENDIX A

Scope of Work

Waste Inventories

The Brookhaven National Laboratory (BNL) shall obtain for the NRC from published documents as complete an inventory as possible of both the commercial and defense generated nuclear waste presently within the U. S. as well as those anticipated to be generated during the next 10 years (1990) and 20 years (2000) within the United States. This inventory shall include all high level waste (including DOE's HLW), commercial spent fuel, commercial transuranic waste and dismantling and decommissioning waste and shall highlight any new and different types of waste. Low level wastes are excluded from this inventory. The inventory shall not be created by BNL but shall be obtained where available from NRC, DOE, their contractors and others as appropriate. However, BNL is responsible for the scope and detail of the inventory and is directed to propose any additional work required to make the inventory complete.

This inventory (in its present form, not in its anticipated disposal form) should include but not be limited to the following: Volume (ft^3 and m^3) and weight (kg, lbs) of waste, its heat generation, its physical form and chemical composition, and the nature and amounts of radionuclides. The latter shall include the curies of total contamination, curies of alpha, curies of beta, curies of gamma, neutron flux, and kilograms of SNM.

APPENDIX B

Glossary of Selected Terms

Actinides

The radioactive elements with atomic number of 89 (i.e., Actinium) and above. The members of the actinide series are therefore ^{89}Ac, ^{90}Th, ^{91}Pa, ^{92}U, ^{93}Np, ^{94}Pu, ^{95}Am, ^{96}Cm, ^{97}Bk, ^{98}Cf, ^{99}Es, ^{100}Fm, ^{101}Md, ^{102}No, and ^{103}Lw.

Activity

A measure of nuclear radiation, given in terms of the number of nuclear disintegrations occurring per second. The standard unit of activity is the curie (Ci), which is 3.7×10^{10} disintegrations per second.

Calcine

A powder or granular form of solid material produced by heating the material to a high temperature (below its melting point) for evaporation of moisture and for oxidation to a chemically stable form. The Idaho high level liquid waste is converted to a calcine.

Commercial Waste

Radioactive waste produced by commercial nuclear power reactors. A small amount of these commercial wastes were reprocessed earlier at West Valley. Most of these wastes are at present stored as spent fuel at the reactor sites.

Decay

The spontaneous transformation of one radionuclide into a different nuclide or into a different energy state of the same nuclide. Every decay process has a definite half life. These are given in Appendix C.

DOE Wastes

Radioactive wastes produced by the U.S. military and research programs. These refer to the wastes generated at the Idaho National Engineering Laboratory, the Savannah River Plant and at the Hanford Reservation.

Diversion Box

An enclosure where inlet and outlet pipes are connected by jumpers, which permits a selection of pathways for liquid process streams or effluents.

Enriched Uranium

Uranium containing more than the natural concentration of the ^{235}U isotope. Natural uranium typically contains 0.7% ^{235}U. Commercial power reactor fuel contains uranium enriched to about 2 to 3%.

Fuel Assembly

A grouping of fuel pins bundled together to form a fuel assembly unit in a reactor core. This group is not taken apart during the loading and unloading of a reactor core.

Half Life

The time in which half the atoms originally present in a given quantity of a radioactive element disintegrate to another element.

High Level Liquid Waste

The raffinate from the first cycle solvent extraction step of a spent fuel reprocessing plant. Equivalent concentrated wastes from subsequent extraction cycles and solvent cleanup sludges are also considered as high level liquid waste.

Ion Exchange

A reversible chemical reaction between a solid and a fluid mixture by means of which ions may be interchanged. The solid material is usually a plastic or a resin. This is used to separate chemicals such as fission products or rare earths from solution.

Production Reactor

A nuclear reactor designed primarily for large scale production of plutonium, tritium or other radionuclides by neutron irradiation.

Purex

A solvent extraction process in which uranium and plutonium are selectively separated from each other and from fission products. This process uses a mixture of tributyl phosphate and a kerosene diluent as the solvent. The Purex process generates an acid solution of high level waste.

Radiolysis

The phenomenon of radiation induced decomposition of chemical compounds. In the high level liquid waste, this causes the decomposition of nitrates to form nitrogen oxides and the decomposition of water molecules to form oxygen and hydrogen.

Raffinate

The liquid part left over after a product has been extracted in a solvent extraction process. This solution is acidic in nature and contains almost all of the fission product activity of the processed material.

Reconstituted Waste

Waste generated by dissolving the salt and mixing with the sludge in the waste tanks for the purpose of feeding to the final solidification facility. The reconstituted waste will be similar in chemical composition to the original fresh waste generated by the fuel reprocessing plant, but will be less radioactive.

Redox

A solvent extraction fuel reprocessing method that uses methylisobutylketone (hexone) as the solvent. The Redox process produces a high level waste with large amounts of aluminum from aluminum nitrate as a salting agent in the solvent extraction process.

Residual Liquor

Liquid left over after formation of salt cake and which cannot be converted to solids by simple evaporation.

Resin

A synthetic organic polymer that can act as an ion exchanger.

Salt Cake

The residue or the product formed (crystallized) after the supernate is evaporated.

Supernate

After the alkaline salt solution resulting from the neutralization of the acidic high level waste settles, the top liquid portion is referred to as the supernate. Other portions of the waste are the insoluble sludge and the crystallized salt. The supernate primarily contains the fission product ^{137}Cs in solution.

Sludge

The precipitate that is formed as a result of neutralization of acidic high level waste. It contains most of the fission products present in the original acidic waste except for cesium, technetium and ruthenium.

Solvent Extraction

A process in which materials are selectively removed from an aqueous solution by contact with an immiscible organic solvent.

Transuranic Elements

Elements with atomic number greater than that of uranium, i.e. greater than 92. They include neptunium, plutonium, americium and curium. Actinides include transuranic elements. Most common transuranic elements in the radioactive waste are ^{239}Pu (half life 24,390 years) and ^{241}Am (half life 458 years).

Transuranic Waste (TRU)

Waste that is contaiminated with transuranic nuclides to the extent that the activity of the waste is equal to or greater than 10 nanocuries per gram of waste.

APPENDIX C

Decay characteristics of radionuclides relevant to nuclear waste management are given in Table C.1. This table contains isotopes which are particularly important to high level radioactive waste management, and therefore many short lived isotopes have been omitted. The list has been compiled from the latest data available in the literature.[1,2] Q-values for the decays are also given, and represent the actual energy or mass difference between the parent and the daughter nuclei in units of MeV. Major radiations are shown in the last column. The maximum and the average electron energy in a beta-spectrum are both given when available; gamma-energy is given for the most intense source, and X-rays and other low energy gamma-rays have been omitted.

Table C.2 shows the principal production modes of long lived isotopes relevant to high level radioactive waste management. The isotopes considered are those that may contribute to the high level waste even after about ten years of storage.

References

1. Table of Isotopes, 7th edition, edited by C. M. Lederer and V. S. Shirley, John Wiley and Sons, Inc., New York (1978); also see The Nuclear Wallet Cards, edited by the same group.

2. A Handbook of Radioactivity Measurements Procedures, NCRP Report No. 58, November 1978.

Table C.1

Decay Characteristics of Radionuclides Relevant to Nuclear Waste Management[1,2]

Nuclide (Z)	Isotope	Half-Life	Principal Mode of Decay[a]	Daughter	Q-Value (MeV)	Major Radiations Mode (MeV) (Intensity)
Tritium (1)	H-3	12.33 y	β	He-3	0.019	β 0.0186 (100%) β_{avg} 0.0057
Carbon (6)	C-14	5730 y	β	N-14	0.156	β 0.156 (100%) β_{avg} 0.0495
Chromium (24)	Cr-51	27.70 d	ε	V-51	0.751	γ 0.32 (10%)
Iron (26)	Fe-55	2.69 y	ε	Mn-55	0.213	
	Fe-59	44.56 d	β	Co-59	1.565	β 0.465 (52.8%), 0.273 (45.6%) β_{avg} 0.117
Nickel (28)	Ni-59	7.5×10^4 y	ε (99+%)	Co-59	1.073	
	Ni-63	100.1 y	β	Cu-63	0.066	β 0.066 (100%) β_{avg} 0.017
Cobalt (27)	Co-60	5.27 y	β (0.25%)	Ni-60	2.824	β 0.318 (99.9%) β_{avg} 0.096
		10.5 m	IT (99.75%)	Co-60		γ 1.173 (100%), 1.332 (100%)
Selenium (34)	Se-79	$\leq 6.5 \times 10^4$ y	β	Br-79	0.159	β 0.160 (100%)
Krypton (36)	Kr-85	10.70 y	β	Rb-85	0.687	β 0.687 (99.57%) β_{avg} 0.251

[a] α alpha decay
β negative beta (negatron) decay
ε positive beta (positron) decay, including EC and/or β^+
IT Isomeric transition (electromagnetic decay from an excited state)

Table C.1 Continued

Nuclide (Z)	Isotope	Half-Life	Principal Mode of Decay[a]	Daughter	Q-Value (MeV)	Major Radiations Mode (MeV) (Intensity)
Rubidium (37)	Rb-86	18.81 d	β (99+%)	Sr-86	1.774	β 1.774 (91.24%), β_{avg} 0.667 γ 1.076 (8.8%)
Strontium (38)	Sr-89	50.55 d	β	Y-89	1.492	β 1.492 (100%), β_{avg} 0.583
	Sr-90	28.82 y	β	Y-90	0.546	β 0.546 (100%), β_{avg} 0.196
Yttrium (39)	Y-90	64.06 h	β	Zr-90	2.284	β 2.284 (100%), β_{avg} 0.935
	Y-91	58.51 d	β	Zr-91	1.543	β 1.543
Zirconium (40)	Zr-93	1.53×10^6 y	β	Nb-93	0.092	β 0.060 (>95%)
	Zr-95	63.98 d	β	Nb-95	1.123	β 0.396 (55%), β 0.360 (43%) γ 0.757
Niobium (41)	Nb-93m	13.63 y	IT	Nb-93		γ 0.030
	Nb-95	34.97 d	β	Mo-95	0.929	β 0.16 (99.92%)
Technetium (43)	Tc-99	2.14×10^5 y	β	Ru-99	0.294	β 0.292
Ruthenium (44)	Ru-103	39.35 d	β	Rh-103	0.763	β 0.225 (87.1%), β_{avg} 0.072 γ 0.497 (90.1%)
	Ru-106	336.5 d	β	Rh-106	0.039	β 0.039 (100%)
Rhodium (45)	Rh-106	29.80 s	β	Pd-106	3.541	β 3.53 (68%) γ 0.622
Palladium (46)	Pd-107	6.5×10^6 y	β	Ag-107	0.033	β 0.04 (100%)
Silver (47)	Ag-110m	252.2 d	β (98.5%) IT (1.5%)	Cd-110	2.893	β 0.084 (67.5%), β 0.531 (30.6%) β_{avg} 0.066, γ 0.658 (94.7%)

Table C.1 Continued

Nuclide (Z)	Isotope	Half-Life	Principal Mode of Decay[a]	Daughter	Q-Value (MeV)	Major Radiations Mode (MeV) (Intensity)
Cadmium (48)	Cd-113m	14.65 y	β (99.9%) IT (0.1%)	In-113	0.322	β 0.585 (100%) γ with IT 0.264
	Cd-115m	44.8 d	β	In-115	1.448	β 1.62 (97%)
Indium (49)	In-114m	49.51 d	IT (96.7%)	In-114		γ with IT 0.190
			ε (3.3%)	Cd-114	1.444	γ with EC 0.725, 0.558
Tin (50)	Sn-117m	14.03 d	IT	Sn-117		γ with 0.159 (86.4%)
	Sn-119m	250 d	IT	Sn-119		γ 0.024 (16.2%)
	Sn-121m	55 y	β-	Sb-121	0.387	β 0.354 (100%)
	Sn-123	129.0 d	β-	Sb-123	1.397	β 1.42 γ 1.089
	Sn-125	9.63 d	β	Sb-125	2.350	β 2.37 (95%) γ 1.066
	Sn-126	$1x10^5$ y	β	Sb-126	0.378	β 0.25 (100%) γ 0.088
Antimony (51)	Sb-124	60.20 d	β	Te-124	2.905	β 0.612 (52%), β 2.303 (22.6%) β_{avg} 0.381
	Sb-125	2.71 y	β	Te-125	0.767	β 0.30 (45%) γ 0.428
	Sb-126	12.4 d	β	Te-126	3.665	β 1.90 γ 0.666
Tellurium (52)	Te-123m	119.7 d	IT	Te-123		γ 0.159
	Te-125m	58 d	IT	Te-125		γ 0.035
	Te-127m	109 d	IT (97.6%)	Te-127		γ with IT 0.088
			β (2.4%)	I-127		γ with β 0.058
	Te-127	9.35 h	β	I-127	0.694	β 0.695 (99%) γ 0.418

Table C.1 Continued

Nuclide (Z)	Isotope	Half-Life	Principal Mode of Decay[a]	Daughter	Q-Value (MeV)	Major Radiations Mode (MeV) (Intensity)
	Te-129m	33.52 d	IT (63.4%) β (36.6%)	Te-129 I-129		γ with IT 0.106 β 1.604 (31%), β_{avg} 0.567
	Te-129	69.5 m	β	I-129	1.498	β 1.470 (89.3%) β_{avg} 0.522
Iodine (53)	I-129	1.57x10 y	β	Xe-129	0.192	β 0.150 (100%), β_{avg} 0.040
Cesium (55)	Cs-134	2.062 y	β (99%)	Ba-134	2.059	β 0.645 (70%), 0.089 (27%)
	Cs-135	2.95×10^6 y	β	Ba-135	0.205	β 0.21 (100%)
	Cs-137	30.17 y	β	Ba-137	1.173	β 0.512 (94%)
Cerium (58)	Ce-141	32.55 d	β	Pr-141	0.580	β 0.435 (70.5%), 0.580 (29.5%) β_{avg} 0.145
	Ce-144	284.5 d	β	Pr-144	0.318	β 0.316 (75.7%)
Praseodymium (59)	Pr-144	17.30 m	β	Nd-144	2.996	β 2.996 (97.8%), β_{avg} 1.207 γ 0.696
Promethium (61)	Pm-147	12.623 y	β	Sm-147	0.225	β 2.464 (55%), 0.999 (33.3%) β_{avg} 0.725
	Pm-148m	41.29 d	β (95.4%) IT (4.6%)	Sm-148 Pm-148		β 0.407 (53.7%) β_{avg} 0.152 γ 0.55, 0.63
	Pm-148	5.37 d	β	Sm-148	2.464	β 2.48 (50%), 1.02 (40%)
Samarium (62)	Sm-151	87.9 y	β	Eu-151	0.076	β 0.055 (88%) γ 0.022

Table C.1 Continued

Nuclide (Z)	Isotope	Half-Life	Principal Mode of Decay[a]	Daughter	Q-Value (MeV)	Major Radiations Mode (MeV) (Intensity)
Europium (63)	Eu-152	13.23 y	β (27%)	Gd-152	1.819	β 0.69 (51%) β_{avg} 0.299, γ 0.344
			EC (73%)	Sm-152	1.877	γ 0.12 - 1.528
	Eu-154	8.55 y	β (99%)	Gd-154	1.978	β 0.590 (45%) β_{avg} 0.234, γ 1.274 (35.5%)
	Eu-155	4.96 y	β	Gd-155	0.246	β 0.152 (84%) β_{avg} 0.045, γ 0.105
Terbium (65)	Tb-158	150 y	EC (82%)	Gd-158	1.216	γ with EC 0.944
			β (18%)	Dy-158	0.935	β 0.853
	Tb-160	72.1 d	β	Dy-160	1.833	β 0.870 (36%)
Thorium (90)	Th-232	1.4×10^{10} y	α	Ra-228	4.081	α 4.016 (77%)
Uranium (92)	U-233	1.59×10^{5} y	α	Th-229	4.909	α 4.824 (84.4%)
	U-234	2.45×10^{5} y	α	Th-230	4.856	α 4.774 (72%)
	U-235	7.04×10^{8} y	α	Th-231	4.679	α 4.397 (57%)
	U-236	2.34×10^{7} y	α	Th-232	4.570	α 4.494 (74%)
	U-238	4.47×10^{9} y	α	Th-234	4.270	α 4.196 (77%)
Neptunium (93)	Np-237	2.14×10^{6} y	α	Pa-233	4.957	α
	Np-239	2.35 d	β	Pu-239	0.721	β 0.438 (48%) β_{avg} 0.118

Table C.1 Continued

Nuclide (Z)	Isotope	Half-Life	Principal Mode of Decay[a]	Daughter	Q-Value (MeV)	Major Radiations Mode (MeV) (Intensity)
Plutonium (94)	Pu-238	87.71 y	α	U-234	5.593	α 5.498 (71.1%)
	Pu-239	2.41×10^4 y	α	U-235	5.244	α 5.155 (73.3%)
	Pu-241	14.36 y	β (99+%)	Am-241	0.021	β .021 (100%)
			α (.002%)	U-237	5.139	α 4.896 (83%)
	Pu-242	3.76×10^5 y	α	U-238	4.893	α 4.901 (74%)
Americium (95)	Am-241	432.02 y	α	Np-237	5.638	α 5.486 (86%)
	Am-242	16.01 h	β (82.7%)	Cm-242	0.661	β 0.625 (49%), γ 0.042
			ε (17.3%)	Pu-242	0.748	γ with EC 0.045
	Am-242m	152 y	IT (99+%)	Am-242		γ with IT 0.049
			α (0.48%)	Np-238	5.585	α 5.205 (89%)
	Am-243	7370 y	α	Np-239	5.439	α 5.276 (87.9%)
Curium (96)	Cm-242	162.76 d	α	Pu-238	6.216	α 6.113 (74%)
	Cm-244	18.10 y	α	Pu-240	5.902	α 5.806 (76.7%)
	Cm-245	8537 y	α	Pu-241	5.623	α 5.362 (93.18%)
	Cm-246	4713 y	α	Pu-242	5.476	α 5.386 (79%)

Table C.2

Principal Means of Production of Several Radioisotopes
Relevant to High Level Waste Management

Isotope	Principal Means of Production[a]
$_{34}Se^{79}$	Fission; $\{_{31}Ga^{79}(\beta)_{32}Ge^{79}(\beta)_{33}As^{79}(\beta)\}$
$_{38}Sr^{90}$	Fission; $\{_{35}Br^{90}(\beta)_{36}Kr^{90}(\beta)_{37}Rb^{90}(\beta)\}$
$_{40}Zr^{93}$	Fission; $\{_{36}Kr^{93}(\beta)_{37}Rb^{93}(\beta)_{38}Sr^{93}(\beta)_{39}Y^{93}(\beta)\}$
$_{43}Tc^{99}$	Fission; $\{_{37}Rb^{99}(\beta)_{38}Sr^{99}(\beta)_{39}Y^{99}(\beta)_{40}Zr^{99}(\beta)_{41}Nb^{99}(\beta)_{42}Mo^{99}(\beta)\}$; $_{42}Mo^{98}(n,\gamma)_{42}Mo^{99}(\beta)$
$_{44}Ru^{106}$	Fission; $\{_{41}Nb^{106}(\beta)_{42}Mo^{106}(\beta)_{43}Tc^{106}(\beta)\}$
$_{45}Rh^{106}$	Fission; $\{_{41}Nb^{106}(\beta)_{42}Mo^{106}(\beta)_{43}Tc^{106}(\beta)_{44}Ru^{106}(\beta)\}$
$_{46}Pd^{107}$	Fission; $\{_{42}Mo^{107}(\beta)_{43}Tc^{107}(\beta)_{44}Ru^{107}(\beta)_{45}Rh^{107}(\beta)\}$
$_{50}Sn^{126}$	Fission; $\{_{49}In^{126}(\beta)\}$
$_{51}Sb^{126}$	Fission; $\{_{49}In^{126}(\beta)_{50}Sn^{126}(\beta)\}$; $_{52}Te^{128}(d,\alpha)$, $_{52}Te^{126}(n,p)$; $_{50}Sn^{124}(\alpha,pn)$
$_{53}I^{129}$	Fission; $\{_{49}In^{129}(\beta)_{50}Sn^{129}(\beta)_{51}Sb^{129}(\beta)_{52}Te^{129}(\beta)\}$
$_{55}Cs^{134}$	$Cs^{133}(n,\gamma)$
$_{55}Cs^{137}$	Fission; $\{_{52}Te^{137}(\beta)_{53}I^{137}(\beta)_{54}Xe^{137}(\beta)\}$
$_{58}Ce^{144}$	Fission; $\{_{54}Xe^{144}(\beta)_{55}Cs^{144}(\beta)_{56}Ba^{144}(\beta)_{57}La^{144}(\beta)\}$
$_{59}Pr^{144}$	Fission; $\{_{54}Xe^{144}(\beta)_{55}Cs^{144}(\beta)_{56}Ba^{144}(\beta)_{57}La^{144}(\beta)_{58}Ce^{144}(\beta)\}$
$_{61}Pm^{147}$	Fission; $\{_{57}La^{147}(\beta)_{58}Ce^{147}(\beta)_{59}Pr^{147}(\beta)_{60}Nd^{147}(\beta)\}$; $_{60}Nd^{146}(n,\gamma)_{60}Nd^{147}(\beta)$
$_{62}Sm^{151}$	Fission; $\{_{58}Ce^{151}(\beta)_{59}Pr^{151}(\beta)_{60}Nd^{151}(\beta)_{61}Pm^{151}(\beta)\}$; $_{62}Sm^{150}(n,\gamma)$
$_{63}Eu^{152}$	$_{63}Eu^{151}(n,\gamma)$
$_{63}Eu^{154}$	$_{63}Eu^{153}(n,\gamma)$
$_{92}U^{233}$	$_{90}Th^{232}(n,\gamma)_{90}Th^{233}(\beta)_{91}Pa^{233}(\beta)$; $_{94}Pu^{233}(Ec)_{93}Np^{233}(Ec)$; $_{93}Np^{237}(\alpha)$ $_{91}Pa^{233}(\beta)$; $_{94}Pu^{237}(\alpha)$; $_{95}Am^{237}(\alpha)_{93}Np^{233}(Ec)$

Table C.2 Continued

Isotope	Principal Means of Production[a]
$_{93}Np^{237}$	$_{92}U^{238}(n,2n)_{92}U^{237}(\beta)$; $_{91}Pa^{237}(\beta)_{92}U^{237}(\beta)$; $_{94}Pu^{241}(\alpha)_{92}U^{237}(\beta)$; $_{95}Am^{241}(\alpha)$; $_{96}Cm^{241}(\alpha)_{94}Pu^{237}(Ec)$; $_{95}Am^{237}(Ec)_{94}Pu^{237}(Ec)$
$_{93}Np^{239}$	$_{92}U^{238}(n,\gamma)_{92}U^{239}(\beta)$; $_{95}Am^{243}(\alpha)$
$_{94}Pu^{238}$	$_{93}Np^{237}(n,\gamma)_{93}Np^{238}(\beta)$; $_{96}Cm^{242}(\alpha)$; $_{98}Cf^{242}(\alpha)_{96}Cm^{238}(Ec)_{95}Am^{238}(Ec)$
$_{94}Pu^{239}$	$_{92}U^{238}(n,\gamma)_{92}U^{239}(\beta)_{93}Np^{239}(\beta)$; $_{96}Cm^{243}(\alpha)$; $_{98}Cf^{243}(\alpha)_{96}Cm^{239}(Ec)$ $_{95}Am^{239}(Ec)$
$_{94}Pu^{240}$	$_{92}U^{240}(\beta)_{93}Np^{240}(\beta)$; $_{96}Cm^{244}(\alpha)$; $_{95}Am^{240}(Ec)$
$_{94}Pu^{241}$	$_{93}Np^{241}(\beta)$; $_{96}Cm^{245}(\alpha)$
$_{95}Am^{241}$	$_{94}Pu^{241}(\beta)$; $_{96}Cm^{241}(Ec)$
$_{96}Cm^{244}$	$_{95}Am^{244}(\beta)$; $_{98}Cf^{248}(\alpha)$; $_{97}Bk^{244}(Ec)$

[a]Cumulative decay of independent fission yields, where applicable, are shown by the mass chain inside { }. Each member of this chain also contributes to the isotope production.

APPENDIX D

Computer Programs Employed in Radioactive Waste Inventory Assessments

DISFUL - A computer program and associated data base developed by S.M. Stoller Corporation.[1] This code may be used to estimate yearly fuel discharges and analyze spent fuel shipping and storage requirememts for various storage scenarios including trans-shipment, pool expansion, and away-from-reactor facilities.

KWIKPLAN - A computer program developed by Oak Ridge National Laboratory.[2] This code is used to calculate the mass flows in the nuclear fuel cycle. Given data on installed nuclear capacity and reference reactor operating data, the code calculates the requirements on uranium milling, fuel reprocessing, and fuel fabrication.

ORIGEN - An isotope generation and depletion code originally developed at Oak Ridge National Laboratory.[3] ORIGEN can calculate the detailed isotopic composition of irradiated fuel elements or stored radioactive materials as a function of time. This code or a modified version of the code has been used extensively in radioactive inventory calculations. A version of this code[4] from the Sandia Laboratory is also operational at Brookhaven.

SWIMS - Solid Waste Information Management System.[5] A computer data management and information system developed and maintained by the Department of Energy for eight solid waste forms. These are TRU, uranium/thorium, fission product, induced activity, tritium, beta-gamma TRU, alpha, and other.

TAFI-II - A depletion calculation code,[6] similar to ORIGEN. This is a Savannah River Laboratory code and is used for their high level waste inventory calculations.

References

1. Spent Fuel Storage Requirements - The Need for Away-From-Reactor Storage, U.S. Department of Energy, DOE/ET-0075, February 1979.

2. Updated Projections of Radioactive Wastes to be Generated by the Nuclear Power Industry, Oak Ridge National Laboratory, ORNL/TM-5427, December 1976.

3. M.J. Bell, "ORIGEN - The ORNL Isotope Generation and Depletion Code," ORNL-4628, May 1973.

4. David E. Bennett, "SANDIA-ORIGEN User's Manual," NUREG/CR-0987 (SAND 79-0299), October 1979.

5. J. E. Dieckenhoner, "Sources, Production Rates and Characteristics of ERDA Low-Level Wastes," Management of Low-Level Radioactive Waste, Volume 1, Pergamon Press Inc., 1979.

6. C. R. Tharin, "TAFI-II, A Computer Code for Management of Radioactive Waste," SRL report DP-966, August 1965.

APPENDIX E

Characteristics of Commercial Spent Fuel

The following tables are provided as a reference for the characteristics of commercial spent fuel from PWR and BWR power reactors. These tables were extracted from "Projections of Spent Fuel to be Discharged by the U. S. Nuclear Power Industry," ORNL/TM-6008.

Table 2.2. Physical characteristics of LWR fuel assemblies

	BWR	PWR
Overall assembly length, m	4.470	4.059
Cross section, cm	13.9 × 13.9	21.4 × 21.4
Fuel element length, m	4.064	3.851
Active fuel height, m	3.759	3.658
Fuel element OD, cm	1.252	1.07
Fuel element array	8 × 8	15 × 15
Assembly total weight, kg	275.7	657.9
Uranium/assembly, kg	183.3	461.4
UO_2/assembly, kg	208.0	523.4
Zircaloy/assembly, kg	57.9[a]	108.4[b]
Hardware/assembly, kg	9.77[c]	26.1[d]
Total metal/assembly, kg	67.7	134.5
Nominal volume/assembly, m^3	0.0864[e]	0.186[e]

[a]Includes Zircaloy fuel-element spacers.

[b]Includes Zircaloy control-rod guide thimbles.

[c]Includes stainless steel tie-plates and Inconel springs.

[d]Includes 10 kg stainless steel nozzles and Inconel-718 grids.

[e]Based on overall outside dimension.

Table 2.3. Total grams charged to a BWR fuel assembly[a]

Elements or isotopes	Grams charged
N	2.555×10^{1}
O	2.474×10^{4}
Si	9.491×10^{1}
Cr	1.986×10^{3}
Mn	1.865×10^{2}
Fe	6.644×10^{3}
Ni	1.157×10^{3}
Zr	1.012×10^{5}
Sn	1.654×10^{3}
Gd	2.867×10^{2}[b]
^{235}U	5.041×10^{3}
^{238}U	1.782×10^{5}

[a]Impurities of 25 ppm of nitrogen and 1 ppm of lithium, in addition to others, are found in the fuel.

[b]Corrected by telephone conversation between C. W. Alexander, ORNL and J. P. Indusi, BNL, 6/26/80. Previously reported as 1.38×10^{5}.

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Table 2.4. Grams of fission-product elements in a BWR assembly

	DISCHARGE	1. YR	2. YR	5. YR	10. YR	30. YR	100. YR	300. YR	1000. YR	3000. YR	10000. YR	30000. YR
H	8.590E-03	8.120E-03	7.675E-03	6.481E-03	4.690E-03	1.584E-03	3.065E-05	3.906E-10	2.883E-27	0.0	0.0	0.0
LI	3.052E-05	3.052E-05	3.052E-05	3.052E-05	3.052E-05	3.052E-05	3.052E-05	3.052E-05	3.052E-05	3.052E-05	3.052E-05	3.052E-05
BE	2.281E-05	2.281E-05	2.281E-05	2.281E-05	[illegible]-05	2.281E-05	2.281E-05	2.281E-05	2.281E-05	2.280E-05	2.276E-05	2.265E-05
C	4.011E-06	4.011E-06	4.010E-06	4.009E-06	4.006E-06	3.997E-06	3.963E-06	3.868E-06	3.554E-06	2.790E-06	1.197E-06	1.065E-07
CO	1.719E-13	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
NI	6.332E-11	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
CU	5.783E-10	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
ZN	8.322E-06	5.711E-09	5.711E-09	5.711E-09	5.711E-09	5.711E-09	5.711E-09	5.711E-09	5.711E-09	5.711E-09	5.711E-09	5.711E-09
GA	4.624E-06	1.379E-07	1.379E-07	1.379E-07	1.379E-07	1.379E-07	1.379E-07	1.379E-07	1.379E-07	1.379E-07	1.379E-07	1.379E-07
GE	9.848E-02	9.843E-02	9.843E-02	9.843E-02	9.843E-02	9.843E-02	9.843E-02	9.843E-02	9.843E-02	9.843E-02	9.843E-02	9.843E-02
AS	3.011E-02	2.979E-02	2.979E-02	2.979E-02	2.979E-02	2.979E-02	2.979E-02	2.979E-02	2.979E-02	2.979E-02	2.979E-02	2.979E-02
SE	8.566E 00	8.566E 00	8.566E 00	8.566E 00	8.566E 00	8.566E 00	8.565E 00	8.563E 00	8.557E 00	8.538E 00	8.476E 00	8.322E 00
BR	3.312E 00	3.311E 00	3.311E 00	3.311E 00	3.311E 00	3.311E 00	3.311E 00	3.313E 00	3.320E 00	3.339E 00	3.401E 00	3.555E 00
KR	5.602E 01	5.578E 01	5.557E 01	5.501E 01	5.429E 01	5.292E 01	5.241E 01	5.240E 01	5.240E 01	5.240E 01	5.240E 01	5.240E 01
RB	5.189E 01	5.212E 01	5.233E 01	5.289E 01	5.361E 01	5.498E 01	5.549E 01	5.550E 01	5.550E 01	5.550E 01	5.550E 01	5.550E 01
SR	1.378E 02	1.323E 02	1.303E 02	1.248E 02	1.165E 02	9.191E 01	6.018E 01	5.337E 01	5.332E 01	5.332E 01	5.332E 01	5.332E 01
Y	7.123E 01	6.941E 01	6.937E 01	6.936E 01	6.936E 01	6.936E 01	6.935E 01	6.935E 01	6.935E 01	6.935E 01	6.935E 01	6.915E 01
ZR	5.387E 02	5.372E 02	5.390E 02	5.445E 02	5.528E 02	5.774E 02	6.091E 02	6.159E 02	6.159E 02	6.157E 02	6.152E 02	6.136E 02
NB	4.897E 00	2.135E-01	5.092E-03	6.505E-04	1.051E-03	2.651E-03	8.253E-03	2.426E-02	8.025E-02	2.401E-01	7.976E-01	2.375E 00
MO	4.948E 02	5.080E 02	5.084E 02	5.084E 02	5.084E 02	5.084E 02	5.084E 02	5.084E 02	5.084E 02	5.084E 02	5.084E 02	5.084E 02
TC	1.199E 02	1.203E 02	1.203E 02	1.203E 02	1.203E 02	1.203E 02	1.203E 02	1.202E 02	1.199E 02	1.191E 02	1.164E 02	1.091E 02
RU	3.604E 02	3.429E 02	3.370E 02	3.318E 02	3.311E 02	3.311E 02	3.311E 02	3.312E 02	3.315E 02	3.322E 02	3.349E 02	3.423E 02
RH	6.855E 01	7.429E 01	7.430E 01	7.430E 01	7.430E 01	7.430E 01	7.430E 01	7.430E 01	7.430E 01	7.430E 01	7.430E 01	7.430E 01
PD	1.851E 02	1.969E 02	2.027E 02	2.079E 02	2.086E 02	2.086E 02	2.086E 02	2.086E 02	2.086E 02	2.086E 02	2.086E 02	2.085E 02
AG	1.228E 01	1.217E 01	1.210E 01	1.213E 01	1.212E 01	1.213E 01	1.213E 01	1.213E 01	1.213E 01	1.214E 01	1.216E 01	1.223E 01
CD	1.533E 01	1.545E 01	1.547E 01	1.548E 01	1.548E 01	1.547E 01	1.546E 01	1.546E 01	1.546E 01	1.546E 01	1.546E 01	1.546E 01
IN	3.611E-01	3.785E-01	3.802E-01	3.846E-01	3.907E-01	4.046E-01	4.131E-01	4.134E-01	4.134E-01	4.134E-01	4.134E-01	4.134E-01
SN	1.311E 01	1.298E 01	1.296E 01	1.296E 01	1.296E 01	1.296E 01	1.295E 01	1.295E 01	1.293E 01	1.287E 01	1.268E 01	1.219E 01
SB	4.545E 00	4.164E 00	3.835E 00	3.201E 00	2.800E 00	2.645E 00	2.644E 00	2.644E 00	2.644E 00	2.644E 00	2.644E 00	2.644E 00
TE	7.294E 01	7.229E 01	7.262E 01	7.325E 01	7.365E 01	7.381E 01	7.381E 01	7.382E 01	7.384E 01	7.389E 01	7.408E 01	7.457E 01
I	3.657E 01	3.592E 01	3.594E 01	3.594E 01	3.594E 01	3.594E 01	3.594E 01	3.594E 01	3.594E 01	3.594E 01	3.593E 01	3.591E 01
XE	8.015E 02	8.017E 02	8.017E 02	8.017E 02	8.017E 02	8.017E 02	8.017E 02	8.017E 02	8.017E 02	8.017E 02	8.017E 02	8.017E 02
CS	4.274E 02	4.204E 02	4.131E 02	3.964E 02	3.764E 02	3.226E 02	2.496E 02	2.316E 02	2.314E 02	2.314E 02	2.313E 02	2.309E 02
BA	2.123E 02	2.183E 02	2.255E 02	2.422E 02	2.622E 02	3.161E 02	3.890E 02	4.070E 02	4.072E 02	4.072E 02	4.074E 02	4.077E 02
LA	1.863E 02	1.859E 02	1.859E 02	1.859E 02	1.859E 02	1.859E 02	1.859E 02	1.859E 02	1.859E 02	1.859E 02	1.859E 02	1.859E 02
CE	4.164E 02	3.833E 02	3.713E 02	3.635E 02	3.630E 02	3.630E 02	3.630E 02	3.630E 02	3.630E 02	3.630E 02	3.630E 02	3.630E 02
PR	1.669E 02	1.710E 02	1.710E 02	1.710E 02	1.710E 02	1.710E 02	1.710E 02	1.710E 02	1.710E 02	1.710E 02	1.710E 02	1.710E 02
ND	5.616E 02	5.927E 02	6.087E 02	6.125E 02	6.130E 02	6.131E 02	6.131E 02	6.130E 02	6.130E 02	6.130E 02	6.130E 02	6.130E 02
PM	2.289E 01	1.784E 01	1.370E 01	6.201E 00	1.654E 00	8.394E-03	7.777E-11	0.0	0.0	0.0	0.0	0.0
SM	1.116E 02	1.175E 02	1.216E 02	1.291E 02	1.335E 02	1.349E 02	1.342E 02	1.333E 02	1.331E 02	1.331E 02	1.331E 02	1.331E 02
EU	2.396E 01	2.303E 01	2.241E 01	2.092E 01	1.926E 01	1.712E 01	1.738E 01	1.817E 01	1.842E 01	1.842E 01	1.842E 01	1.842E 01
GD	1.138E 01	1.245E 01	1.308E 01	1.462E 01	1.636E 01	1.879E 01	1.930E 01	1.931E 01	1.931E 01	1.931E 01	1.931E 01	1.931E 01
TB	3.993E-01	3.890E-01	3.888E-01	3.887E-01	3.887E-01	3.887E-01	3.887E-01	3.887E-01	3.887E-01	3.887E-01	3.887E-01	3.887E-01
DY	1.939E-01	2.049E-01	2.052E-01	2.052E-01	2.052E-01	2.052E-01	2.052E-01	2.052E-01	2.052E-01	2.052E-01	2.052E-01	2.052E-01
HO	1.930E-02	1.929E-02	1.929E-02	1.929E-02	1.929E-02	1.928E-02	1.928E-02	1.926E-02	1.921E-02	1.918E-02	1.911E-02	1.911E-02
ER	7.490E-03	7.529E-03	7.529E-03	7.530E-03	7.530E-03	7.532E-03	7.539E-03	7.557E-03	7.607E-03	7.674E-03	7.705E-03	7.706E-03
TM	7.393E-06	7.005E-06	6.891E-06	6.853E-06	6.844E-06	6.842E-06	6.842E-06	6.842E-06	6.842E-06	6.842E-06	6.842E-06	6.842E-06
YB	1.090E-06	1.808E-06	1.921E-06	1.959E-06	1.968E-06	1.970E-06	1.970E-06	1.970E-06	1.970E-06	1.970E-06	1.970E-06	1.970E-06
TOTAL	5.199E 03	5.199E 03	5.199E 03	5.199E 03	5.199E 03	5.199E 03	5.199E 03	5.199E 03	5.199E 03	5.199E 03	5.199E 03	5.199E 03

Table 2.5. Grams of heavy elements in a BWR assembly

	DISCHARGE	1. YR	2. YR	5. YR	10. YR	30. YR	100. YR	300. YR	1000. YR	3000. YR	10000. YR	30000. YR
HE	4.414E-02	7.391E-02	8.460E-02	1.034E-01	1.351E-01	2.802E-01	8.082E-01	1.974E 00	4.227E 00	6.835E 00	1.188E 01	1.904E 01
SP	2.214E-07	4.403E-07	5.800E-07	9.186E-07	1.398E-06	2.637E-06	3.815E-06	4.512E-06	6.585E-06	1.188E-05	2.529E-05	4.558E-05
TL	2.593E-13	5.741E-13	8.799E-13	1.610E-12	2.191E-12	2.117E-12	1.111E-12	2.598E-13	3.459E-13	1.014E-12	3.387E-12	9.822E-12
PB	6.739E-08	2.024E-07	4.396E-07	1.684E-06	4.876E-06	1.958E-05	5.483E-05	8.850E-05	1.663E-04	1.642E-03	3.303E-02	3.832E-01
BI	2.451E-11	4.703E-11	6.833E-11	1.229E-10	1.827E-10	3.561E-10	2.574E-09	5.016E-08	2.347E-06	8.800E-05	3.593E-03	7.199E-02
PO	3.311E-14	6.661E-14	1.040E-13	3.399E-13	1.585E-12	2.292E-11	4.980E-10	6.908E-09	9.771E-08	7.202E-07	4.157E-06	1.453E-05
AT	1.973E-20	6.577E-21	6.823E-21	7.646E-21	9.242E-21	1.846E-20	9.064E-20	7.608E-19	1.180E-17	1.439E-16	1.589E-15	9.219E-15
RN	2.485E-13	5.404E-13	8.288E-13	1.555E-12	2.291E-12	4.271E-12	2.655E-11	2.490E-10	2.859E-09	2.108E-08	1.216E-07	4.252E-07
FR	2.097E-16	1.194E-16	1.518E-16	2.484E-16	4.074E-16	1.032E-15	3.383E-15	1.413E-14	1.321E-13	1.389E-12	1.479E-11	8.513E-11
RA	4.338E-09	8.091E-09	1.248E-08	2.880E-08	6.665E-08	3.858E-07	3.992E-06	3.875E-05	4.453E-04	3.282E-03	1.895E-02	6.623E-02
AC	1.083E-09	2.221E-09	3.352E-09	6.695E-09	1.212E-08	3.239E-08	9.582E-08	2.688E-07	8.992E-07	2.617E-06	8.518E-06	2.378E-05
TH	2.482E-04	3.453E-04	4.430E-04	7.384E-04	1.238E-03	3.322E-03	1.151E-02	3.883E-02	1.423E-01	4.392E-01	1.465E 00	4.162E 00
PA	5.823E-05	5.973E-05	6.106E-05	6.503E-05	7.167E-05	9.825E-05	1.915E-04	4.574E-04	1.384E-03	4.009E-03	1.298E-02	3.602E-02
U	1.765E 05	1.765E 05	1.765E 05	1.765E 05	1.765E 05	1.765E 05	1.765E 05	1.765E 05	1.765E 05	1.767E 05	1.769E 05	1.773E 05
NP	7.335E 01	6.307E 01	6.311E 01	6.328E 01	6.380E 01	6.771E 01	8.740E 01	1.366E 02	2.244E 02	2.650E 02	2.661E 02	2.644E 02
PU	1.546E 03	1.550E 03	1.541E 03	1.516E 03	1.482E 03	1.403E 03	1.346E 03	1.325E 03	1.280E 03	1.170E 03	8.795E 02	4.673E 02
AM	1.840E 01	2.760E 01	3.651E 01	6.055E 01	9.345E 01	1.659E 02	1.924E 02	1.439E 02	5.390E 01	1.067E 01	4.727E 00	7.719E-01
CM	4.279E 00	2.718E 00	2.318E 00	1.996E 00	1.655E 00	7.889E-01	8.022E-02	2.401E-02	2.212E-02	1.784E-02	8.654E-03	1.341E-03
BK	8.122E-09	3.684E-09	1.670E-09	1.556E-10	2.978E-12	3.999E-19	8.719E-24	8.650E-24	8.412E-24	7.767E-24	5.877E-24	2.650E-24
CF	3.215E-09	7.512E-09	9.385E-09	1.054E-08	1.024E-08	9.174E-09	7.692E-09	5.213E-09	1.361E-09	4.299E-11	8.280E-14	1.112E-19
ES	3.806E-13	3.159E-15	1.255E-15	7.986E-17	8.102E-19	8.559E-27	0.0	0.0	0.0	0.0	0.0	0.0
TOTAL	1.781E 05	1.781E 05	1.781E 05	1.781E 05	1.781E 05	1.781E 05	1.781E 05	1.781E 05	1.781E 05	1.781E 05	1.781E 05	1.781E 05

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Table 2.6. Curies of light-element isotopes in a BWR assembly

	DISCHARGE	1. YR	2. YR	5. YR	10. YR	30. YR	100. YR	300. YR	1000. YR	3000. YR	10000. YR	30000. YR
H 3	4.752E 01	4.491E 01	4.245E 01	3.585E 01	2.705E 01	8.763E 00	1.696E-01	2.161E-06	1.595E-23	0.0	0.0	0.0
C 14	2.634E-01	2.633E-01	2.633E-01	2.632E-01	2.631E-01	2.624E-01	2.602E-01	2.580E-01	2.338E-01	1.832E-01	7.856E-02	6.990E-03
CL 36	1.968E-03	1.968E-03	1.968E-03	1.967E-03	1.967E-03	1.967E-03	1.967E-03	1.966E-03	1.963E-03	1.954E-03	1.924E-03	1.840E-03
MN 54	2.467E 01	1.070E 01	4.639E 00	3.782E-01	5.799E-03	3.205E-10	0.0	0.0	0.0	0.0	0.0	0.0
FE 55	2.152E 02	1.648E 02	1.263E 02	5.675E 01	1.496E 01	7.228E-02	5.680E-10	0.0	0.0	0.0	0.0	0.0
CO 58	1.588E 02	8.557E 00	1.308E-01	3.092E-06	6.023E-14	0.0	0.0	0.0	0.0	0.0	0.0	0.0
CO 60	4.179E 02	3.663E 02	3.211E 02	2.162E 02	1.119E 02	8.020E 00	7.908E-04	2.032E-15	0.0	0.0	0.0	0.0
NI 59	1.508E-01	1.508E-01	1.508E-01	1.508E-01	1.508E-01	1.507E-01	1.506E-01	1.504E-01	1.495E-01	1.469E-01	1.383E-01	1.163E-01
NI 63	2.297E 01	2.280E 01	2.263E 01	2.212E 01	2.130E 01	1.832E 01	1.081E 01	2.396E 00	1.227E-02	3.505E-09	0.0	0.0
ZN 65	2.020E 01	7.189E 00	2.558E 00	1.152E-01	6.577E-04	6.957E-13	0.0	0.0	0.0	0.0	0.0	0.0
ZR 93	9.300E-02	9.300E-02	9.300E-02	9.300E-02	9.300E-02	9.300E-02	9.299E-02	9.298E-02	9.293E-02	9.280E-02	9.232E-02	9.099E-02
ZR 95	1.508E 04	3.160E 02	6.622E 00	6.094E-05	2.463E-13	0.0	0.0	0.0	0.0	0.0	0.0	0.0
NB 93M	7.032E-03	1.160E-02	1.591E-02	2.744E-02	4.273E-02	7.399E-02	8.809E-02	8.833E-02	8.829E-02	8.816E-02	8.771E-02	8.644E-02
NB 94	5.889E-02	5.889E-02	5.889E-02	5.888E-02	5.887E-02	5.883E-02	5.869E-02	5.828E-02	5.689E-02	5.307E-02	4.163E-02	2.081E-02
NB 95	1.500E 04	6.681E 02	1.468E 01	1.314E-04	5.310E-13	0.0	0.0	0.0	0.0	0.0	0.0	0.0
NB 95M	1.869E 02	4.013E 00	8.410E-02	7.739E-07	3.127E-15	0.0	0.0	0.0	0.0	0.0	0.0	0.0
MO 93	1.434E-03	1.433E-03	1.433E-03	1.432E-03	1.430E-03	1.424E-03	1.401E-03	1.338E-03	1.138E-03	7.168E-04	1.422E-04	1.398E-06
IN113M	2.895E 02	3.204E 01	3.544E 00	4.919E-03	8.155E-08	6.158E-27	0.0	0.0	0.0	0.0	0.0	0.0
SN113	2.894E 02	3.202E 01	3.542E 00	4.916E-03	8.150E-08	6.155E-27	0.0	0.0	0.0	0.0	0.0	0.0
SN119M	2.224E 03	7.915E 02	2.817E 02	1.269E 01	7.243E-02	7.676E-11	0.0	0.0	0.0	0.0	0.0	0.0
SN121M	2.430E-01	2.397E-01	2.364E-01	2.267E-01	2.115E-01	1.603E-01	6.071E-02	3.789E-03	2.300E-07	2.060E-19	0.0	0.0
SN123	1.054E 02	1.482E 01	2.084E 00	5.792E-03	3.183E-07	2.903E-24	0.0	0.0	0.0	0.0	0.0	0.0
SB125	6.318E 02	4.958E 02	3.843E 02	1.794E 02	5.041E 01	3.147E-01	6.012E-09	0.0	0.0	0.0	0.0	0.0
TE125M	1.369E 02	1.207E 02	9.385E 01	4.382E 01	1.231E 01	7.685E-02	1.468E-09	0.0	0.0	0.0	0.0	0.0
EU154	2.830E 01	2.611E 01	2.409E 01	1.891E 01	1.263E 01	2.517E 00	8.860E-03	8.744E-10	0.0	0.0	0.0	0.0
EU155	1.579E 01	1.367E 01	1.183E 01	7.668E 00	3.723E 00	2.070E-01	8.390E-06	2.369E-18	0.0	0.0	0.0	0.0
GD153	2.577E 02	9.004E 01	3.146E 01	1.381E 00	7.004E-03	5.140E-12	0.0	0.0	0.0	0.0	0.0	0.0
TB160	1.194E 03	3.601E 01	1.086E 00	2.975E-05	7.815E-13	0.0	0.0	0.0	0.0	0.0	0.0	0.0
SUMTOT	3.635E 04	3.263E 03	1.379E 03	5.961E 02	2.552E 02	3.910E 01	1.171E 01	3.047E 00	6.363E-01	5.668E-01	4.805E-01	3.233E-01
TOTAL	1.175E 05	3.267E 03	1.380E 03	5.962E 02	2.552E 02	3.910E 01	1.171E 01	3.048E 00	6.367E-01	5.671E-01	4.808E-01	3.236E-01

Table 2.7. Curies of fission-product isotopes in a BWR assembly

	DISCHARGE	1. YR	2. YR	5. YR	10. YR	30. YR	100. YR	300. YR	1000. YR	3000. YR	10000. YR	30000. YR
H 3	8.326E 01	7.870E 01	7.439E 01	6.282E 01	4.739E 01	1.535E 01	2.971E-01	3.786E-06	2.795E-23	0.0	0.0	0.0
SE 79	6.212E-02	6.212E-02	6.212E-02	6.212E-02	6.212E-02	6.211E-02	6.206E-02	6.193E-02	6.147E-02	6.017E-02	5.584E-02	4.511E-02
KR 85	1.414E 03	1.326E 03	1.243E 03	1.024E 03	7.415E 02	2.037E 02	2.213E 00	5.417E-06	1.243E-25	0.0	0.0	0.0
SR 89	9.903E 04	7.612E 02	5.853E 00	2.657E-06	7.122E-17	0.0	0.0	0.0	0.0	0.0	0.0	0.0
SR 90	1.144E 04	1.116E 04	1.089E 04	1.011E 04	8.939E 03	5.458E 03	9.709E 02	6.989E 00	2.217E-07	8.335E-29	0.0	0.0
Y 90	1.183E 04	1.116E 04	1.089E 04	1.011E 04	8.941E 03	5.459E 03	9.711E 02	6.995E 00	2.217E-07	8.337E-29	0.0	0.0
Y 91	1.301E 05	1.741E 03	2.315E 01	5.440E-05	2.259E-14	0.0	0.0	0.0	0.0	0.0	0.0	0.0
ZR 93	4.438E-01	4.440E-01	4.440E-01	4.440E-01	4.440E-01	4.440E-01	4.440E-01	4.439E-01	4.437E-01	4.430E-01	4.408E-01	4.344E-01
NB 93M	3.707E-02	5.868E-02	7.908E-02	1.337E-01	2.060E-01	3.539E-01	4.206E-01	4.217E-01	4.215E-01	4.209E-01	4.188E-01	4.127E-01
ZR 95	1.883E 05	3.946E 03	8.270E 01	7.610E-04	3.075E-12	0.0	0.0	0.0	0.0	0.0	0.0	0.0
NB 95	1.911E 05	8.346E 03	1.833E 02	1.641E-03	6.632E-12	0.0	0.0	0.0	0.0	0.0	0.0	0.0
TC 99	2.042E 00	2.050E 00	2.050E 00	2.050E 00	2.050E 00	2.050E 00	2.049E 00	2.048E 00	2.043E 00	2.030E 00	1.984E 00	1.858E 00
RU103	1.889E 05	3.158E 02	5.171E-01	2.415E-09	3.151E-23	0.0	0.0	0.0	0.0	0.0	0.0	0.0
RH103M	1.890E 05	3.161E 02	5.284E-01	2.417E-09	3.154E-23	0.0	0.0	0.0	0.0	0.0	0.0	0.0
RU106	7.810E 04	3.933E 04	1.980E 04	2.528E 03	8.181E 01	8.976E-05	1.242E-25	0.0	0.0	0.0	0.0	0.0
RH106	8.383E 04	3.933E 04	1.980E 04	2.528E 03	8.181E 01	8.976E-05	1.242E-25	0.0	0.0	0.0	0.0	0.0
PD107	1.739E-02	1.739E-02	1.739E-02	1.739E-02	1.739E-02	1.739E-02	1.739E-02	1.739E-02	1.739E-02	1.738E-02	1.737E-02	1.733E-02
SB125	2.064E 03	1.613E 03	1.251E 03	5.842E 02	1.641E 02	1.024E 00	1.956E-08	0.0	0.0	0.0	0.0	0.0
TE125M	4.453E 02	3.931E 02	3.056E 02	1.427E 02	4.008E 01	2.501E-01	4.777E-09	0.0	0.0	0.0	0.0	0.0
SN126	1.158E-01	1.158E-01	1.158E-01	1.158E-01	1.158E-01	1.157E-01	1.157E-01	1.155E-01	1.150E-01	1.134E-01	1.080E-01	9.402E-02
SB126	1.273E 02	1.621E-02	1.621E-02	1.621E-02	1.621E-02	1.620E-02	1.620E-02	1.617E-02	1.610E-02	1.587E-02	1.512E-02	1.316E-02
SB126M	6.310E 01	1.158E-01	1.158E-01	1.158E-01	1.158E-01	1.157E-01	1.157E-01	1.155E-01	1.150E-01	1.134E-01	1.080E-01	9.402E-02
I129	4.742E-03	4.800E-03	4.800E-03	4.800E-03	4.800E-03	4.800E-03	4.800E-03	4.800E-03	4.800E-03	4.800E-03	4.798E-03	4.794E-03
CS134	2.035E 04	1.454E 04	1.039E 04	3.785E 03	7.037E 02	8.771E-01	5.177E-11	0.0	0.0	0.0	0.0	0.0
CS135	6.522E-02	6.532E-02	6.532E-02	6.532E-02	6.532E-02	6.532E-02	6.532E-02	6.531E-02	6.530E-02	6.526E-02	6.512E-02	6.473E-02
CS137	1.577E 04	1.541E 04	1.506E 04	1.406E 04	1.253E 04	7.904E 03	1.577E 03	1.576E 01	1.573E-06	1.568E-26	0.0	0.0
BA137M	1.495E 04	1.458E 04	1.425E 04	1.330E 04	1.185E 04	7.477E 03	1.492E 03	1.491E 01	1.488E-06	1.483E-26	0.0	0.0
CE144	1.584E 05	6.501E 04	2.669E 04	1.847E 03	2.156E 01	3.988E-07	0.0	0.0	0.0	0.0	0.0	0.0
PR144	1.592E 05	6.502E 04	2.669E 04	1.847E 03	2.156E 01	3.988E-07	0.0	0.0	0.0	0.0	0.0	0.0
PR144M	1.902E 03	7.802E 02	3.203E 02	2.216E 01	2.587E-01	4.786E-09	0.0	0.0	0.0	0.0	0.0	0.0
PM147	2.068E 04	1.655E 04	1.271E 04	5.751E 03	1.534E 03	7.785E 00	7.213E-08	0.0	0.0	0.0	0.0	0.0
SM151	5.702E 01	5.743E 01	5.700E 01	5.574E 01	5.370E 01	4.626E 01	2.745E 01	6.176E 00	3.336E-02	1.110E-08	0.0	0.0
EU154	1.499E 03	1.383E 03	1.275E 03	1.001E 03	6.690E 02	1.333E 02	4.653E-01	4.592E-08	0.0	0.0	0.0	0.0
EU155	9.680E 02	8.378E 02	7.251E 02	4.701E 02	2.283E 02	1.269E 01	5.144E-04	1.452E-16	0.0	0.0	0.0	0.0
SUMTOT	1.570E 06	3.140E 05	1.727E 05	6.923E 04	4.665E 04	2.672E 04	5.044E 03	5.414E 01	3.337E 00	3.284E 00	3.218E 00	3.039E 00
TOTAL	1.998E 07	3.149E 05	1.729E 05	6.924E 04	4.666E 04	2.673E 04	5.044E 03	5.414E 01	3.337E 00	3.284E 00	3.218E 00	3.039E 00

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Table 2.8. Curies of heavy-element isotopes in a BWR assembly

	DISCHA E	1. YR	2. YR	5. YR	10. YR	30. YR	100. YR	300. YR	1000. YR	3000. YR	10000. YR	30000. YR
PB210	2.417E-10	3.759E-10	5.737E-10	1.816E-09	7.120E-09	1.033E-07	2.245E-06	3.114E-05	4.404E-04	3.247E-03	1.874E-02	6.550E-02
PB214	2.995E-09	5.132E-09	7.942E-09	2.043E-08	5.497E-08	3.710E-07	3.943E-06	3.833E-05	4.404E-04	3.247E-03	1.874E-02	6.550E-02
BI210	2.428E-10	3.761E-10	5.739E-10	1.816E-09	7.121E-09	1.033E-07	2.245E-06	3.114E-05	4.404E-04	3.247E-03	1.874E-02	6.550E-02
BI214	2.995E-09	5.132E-09	7.942E-09	2.043E-08	5.497E-08	3.710E-07	3.943E-06	3.833E-05	4.404E-04	3.247E-03	1.874E-02	6.550E-02
PO210	1.465E-10	2.942E-10	4.592E-10	1.515E-09	7.121E-09	1.033E-07	2.245E-06	3.114E-05	4.404E-04	3.247E-03	1.874E-02	6.550E-02
PO214	7.683E-09	5.132E-09	7.942E-09	2.043E-08	5.497E-08	3.710E-07	3.943E-06	3.833E-05	4.404E-04	3.247E-03	1.874E-02	6.550E-02
PO218	2.995E-09	5.132E-09	7.942E-09	2.043E-08	5.497E-08	3.710E-07	3.943E-06	3.833E-05	4.404E-04	3.247E-03	1.874E-02	6.550E-02
RN222	2.995E-09	5.132E-09	7.942E-09	2.043E-08	5.497E-08	3.710E-07	3.943E-06	3.833E-05	4.404E-04	3.247E-03	1.874E-02	6.550E-02
RA226	2.993E-09	5.132E-09	7.942E-09	2.043E-08	5.497E-08	3.710E-07	3.943E-06	3.833E-05	4.404E-04	3.247E-03	1.874E-02	6.550E-02
TH230	4.167E-06	5.716E-06	7.273E-06	1.199E-05	2.000E-05	5.378E-05	1.893E-04	6.520E-04	2.416E-03	7.417E-03	2.408E-02	6.509E-02
TH234	5.816E-02	5.812E-02	5.813E-02	5.812E-02	5.812E-02	5.812E-02	5.812E-02	5.812E-02	5.812E-02	5.812E-02	5.812E-02	5.813E-02
PA233	4.247E-02	4.448E-02	4.450E-02	4.462E-02	4.499E-02	4.775E-02	6.163E-02	9.630E-02	1.582E-01	1.868E-01	1.876E-01	1.864E-01
PA234M	5.861E-02	5.812E-02	5.813E-02	5.812E-02	5.812E-02	5.812E-02	5.812E-02	5.812E-02	5.812E-02	5.812E-02	5.812E-02	5.813E-02
U234	1.784E-01	1.792E-01	1.801E-01	1.828E-01	1.870E-01	2.025E-01	2.411E-01	2.833E-01	2.942E-01	2.929E-01	2.883E-01	2.758E-01
U236	3.852E-02	3.852E-02	3.852E-02	3.853E-02	3.858E-02	3.859E-02	3.876E-02	3.925E-02	4.087E-02	4.490E-02	5.398E-02	6.149E-02
U238	5.812E-02	5.812E-02	5.812E-02	5.812E-02	5.812E-02	5.812E-02	5.812E-02	5.812E-02	5.812E-02	5.812E-02	5.812E-02	5.813E-02
NP237	4.366E-02	4.448E-02	4.450E-02	4.462E-02	4.499E-02	4.775E-02	6.163E-02	9.630E-02	1.582E-01	1.868E-01	1.876E-01	1.864E-01
NP239	2.630E 06	2.250E 00	2.250E 00	2.249E 00	2.248E 00	2.244E 00	2.230E 00	2.190E 00	2.055E 00	1.715E 00	9.094E-01	1.485E-01
PU238	2.885E 02	3.130E 02	3.158E 02	3.099E 02	2.981E 02	2.552E 02	1.481E 02	3.140E 01	1.501E-01	1.907E-06	2.581E-20	0.0
PU239	5.361E 01	5.431E 01	5.431E 01	5.430E 01	5.430E 01	5.427E 01	5.417E 01	5.387E 01	5.285E 01	5.004E 01	4.124E 01	2.353E 01
PU240	8.481E 01	8.482E 01	8.483E 01	8.485E 01	8.488E 01	8.490E 01	8.444E 01	8.273E 01	7.700E 01	6.273E 01	3.060E 01	3.936E 00
PU241	2.039E 04	1.945E 04	1.854E 04	1.608E 04	1.268E 04	4.908E 03	1.768E 02	1.582E-02	2.375E-03	2.009E-03	1.117E-03	2.087E-04
PU242	2.708E-01	2.708E-01	2.708E-01	2.708E-01	2.708E-01	2.708E-01	2.708E-01	2.707E-01	2.704E-01	2.698E-01	2.660E-01	2.564E-01
AM241	2.267E 01	5.449E 01	8.477E 01	1.672E 02	2.799E 02	5.285E 02	6.194E 02	4.542E 02	1.481E 02	6.030E 00	1.199E-03	2.087E-04
AM243	2.248E 00	2.250E 00	2.250E 00	2.249E 00	2.248E 00	2.244E 00	2.230E 00	2.190E 00	2.055E 00	1.715E 00	9.094E-01	1.485E-01
CM242	6.199E 03	1.320E 03	2.799E 02	3.308E 00	6.513E-01	5.933E-01	4.311E-01	1.732E-01	7.115E-03	7.808E-07	1.070E-20	0.0
CM244	1.895E 02	1.824E 02	1.756E 02	1.565E 02	1.292E 02	6.009E 01	4.117E 00	1.942E-03	4.428E-15	0.0	0.0	0.0
SUMTOT	2.657E 06	2.146E 04	1.954E 04	1.686E 04	1.354E 04	5.897E 03	1.093E 03	6.277E 02	2.833E 02	1.234E 02	7.501E 01	2.956E 01
TOTAL	5.474E 06	2.146E 04	1.955E 04	1.687E 04	1.354E 04	5.899E 03	1.094E 03	6.282E 02	2.834E 02	1.234E 02	7.506E 01	2.973E 01

Table 2.9. Watts of light-element isotopes in a BWR assembly

	DISCHARGE	1. YR	2. YR	5. YR	10. YR	30. YR	100. YR	300. YR	1000. YR	3000. YR	10000. YR	30000. YR
H 3	1.690E-03	1.597E-03	1.510E-03	1.275E-03	9.620E-04	3.117E-04	6.032E-06	7.685E-11	5.673E-28	0.0	0.0	0.0
C 14	2.437E-04	2.437E-04	2.436E-04	2.436E-04	2.434E-04	2.428E-04	2.408E-04	2.350E-04	2.159E-04	1.695E-04	7.269E-05	6.468E-06
CL 36	8.397E-06	8.397E-06	8.397E-06	8.397E-06	8.397E-06	8.397E-06	8.395E-06	8.392E-06	8.378E-06	8.341E-06	8.211E-06	7.852E-06
MN 54	2.016E-01	8.744E-02	3.792E-02	3.092E-03	4.740E-05	2.620E-12	0.0	0.0	0.0	0.0	0.0	0.0
FE 55	2.960E-01	2.267E-01	1.736E-01	7.804E-02	2.058E-02	9.940E-05	7.811E-13	0.0	0.0	0.0	0.0	0.0
CO 58	2.173E 00	6.237E-02	1.790E-03	4.232E-08	8.243E-16	0.0	0.0	0.0	0.0	0.0	0.0	0.0
CO 60	6.983E 00	6.121E 00	5.365E 00	3.613E 00	1.870E 00	1.340E-01	1.321E-05	4.732E-17	0.0	0.0	0.0	0.0
NI 59	9.598E-04	9.598E-04	9.598E-04	9.598E-04	9.597E-04	9.596E-04	9.590E-04	9.573E-04	9.515E-04	9.352E-04	8.802E-04	7.401E-04
NI 63	9.123E-03	9.054E-03	8.986E-03	8.785E-03	8.461E-03	7.277E-03	4.295E-03	9.517E-04	4.872E-06	1.392E-12	0.0	0.0
ZN 65	1.617E-01	5.753E-02	2.047E-02	9.220E-04	5.263E-06	5.567E-15	0.0	0.0	0.0	0.0	0.0	0.0
ZR 93	1.103E-05	1.103E-05	1.103E-05	1.103E-05	1.103E-05	1.103E-05	1.102E-05	1.102E-05	1.102E-05	1.100E-05	1.095E-05	1.079E-05
ZR 95	7.616E 01	1.596E 00	3.344E-02	3.078E-07	1.244E-15	0.0	0.0	0.0	0.0	0.0	0.0	0.0
NB 93M	1.250E-06	2.063E-06	2.829E-06	4.680E-06	7.599E-06	1.316E-05	1.567E-05	1.571E-05	1.570E-05	1.568E-05	1.560E-05	1.537E-05
NB 94	6.113E-04	6.113E-04	6.112E-04	6.112E-04	6.111E-04	6.106E-04	6.092E-04	6.050E-04	5.904E-04	5.509E-04	4.321E-04	2.160E-04
NB 95	7.192E 01	3.204E 00	7.038E-02	6.302E-07	2.547E-15	0.0	0.0	0.0	0.0	0.0	0.0	0.0
MO 93	3.569E-06	3.568E-06	3.567E-06	3.565E-06	3.561E-06	3.544E-06	3.488E-06	3.330E-06	2.833E-06	1.785E-06	3.581E-07	3.482E-09
AG108M	1.149E-05	1.143E-05	1.137E-05	1.119E-05	1.089E-05	9.791E-06	6.739E-06	2.319E-06	5.537E-08	1.286E-12	7.764E-29	0.0
IN113M	6.744E-01	7.463E-02	8.257E-03	1.146E-05	1.900E-10	1.435E-29	0.0	0.0	0.0	0.0	0.0	0.0
SN113	1.081E 00	1.196E-01	1.323E-02	1.836E-05	3.043E-10	2.298E-29	0.0	0.0	0.0	0.0	0.0	0.0
SN119M	1.173E 00	4.176E-01	1.486E-01	6.695E-03	3.821E-05	4.050E-14	0.0	0.0	0.0	0.0	0.0	0.0
SN121M	4.869E-04	4.802E-04	4.736E-04	4.543E-04	4.238E-04	3.212E-04	1.216E-04	7.591E-06	4.608E-10	4.127E-22	0.0	0.0
SN123	4.979E-01	7.001E-02	9.844E-03	2.736E-05	1.504E-09	1.372E-26	0.0	0.0	0.0	0.0	0.0	0.0
SB125	2.019E 00	1.583E 00	1.228E 00	5.733E-01	1.611E-01	1.006E-03	1.921E-11	0.0	0.0	0.0	0.0	0.0
TE125M	1.169E-01	1.031E-01	8.011E-02	3.740E-02	1.051E-02	6.560E-05	1.253E-12	0.0	0.0	0.0	0.0	0.0
EU154	2.511E-01	2.317E-01	2.137E-01	1.678E-01	1.121E-01	2.234E-02	7.862E-05	7.759E-12	0.0	0.0	0.0	0.0
EU155	1.329E-02	1.150E-02	9.956E-03	6.454E-03	3.134E-03	1.743E-04	7.062E-09	1.994E-21	0.0	0.0	0.0	0.0
GD153	1.833E-01	6.405E-02	2.238E-02	9.542E-04	4.982E-06	3.656E-15	0.0	0.0	0.0	0.0	0.0	0.0
TB160	7.369E 00	2.222E-01	6.698E-03	1.836E-07	4.575E-15	0.0	0.0	0.0	0.0	0.0	0.0	0.0
TA182	1.462E-01	1.617E-02	1.789E-03	2.346E-06	1.576E-09	1.537E-09	1.537E-09	1.537E-09	1.537E-09	1.537E-09	1.536E-09	1.534E-09
SUMTOT	1.714E 02	1.428E 01	7.458E 00	4.500E 00	2.189E 00	1.675E-01	6.368E-03	2.797E-03	1.801E-03	1.692E-03	1.420E-03	9.966E-04
TOTAL	6.080E 02	1.430E 01	7.459E 00	4.500E 00	2.189E 00	1.675E-01	6.370E-03	2.799E-03	1.802E-03	1.693E-03	1.420E-03	9.968E-04

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Table 2.10. Watts of fission-product isotopes in a BWR assembly

	DISCHARGE	1. YR	2. YR	5. YR	10. YR	30. YR	100. YR	300. YR	1000. YR	3000. YR	10000. YR	30000. YR
SE 79	1.547E-05	1.547E-05	1.547E-05	1.547E-05	1.547E-05	1.546E-05	1.545E-05	1.542E-05	1.530E-05	1.498E-05	1.390E-05	1.123E-05
KR 85	2.121E 00	1.989E 00	1.865E 00	1.536E 00	1.112E 00	3.055E-01	3.319E-03	8.124E-09	1.864E-28	0.0	0.0	0.0
SR 89	3.416E 02	2.626E 00	2.019E-02	9.165E-09	2.457E-19	0.0	0.0	0.0	0.0	0.0	0.0	0.0
SR 90	1.343E 01	1.310E 01	1.278E 01	1.187E 01	1.049E 01	6.406E 00	1.139E 00	8.203E-03	2.602E-10	9.782E-32	0.0	0.0
Y 90	6.531E 01	6.161E 01	6.011E 01	5.582E 01	4.934E 01	3.013E 01	5.359E 00	3.860E-02	1.224E-09	4.601E-31	0.0	0.0
Y 91	4.695E 02	6.284E 00	8.356E-02	1.964E-07	8.157E-17	0.0	0.0	0.0	0.0	0.0	0.0	0.0
ZR 93	5.262E-05	5.264E-05	5.264E-05	5.264E-05	5.264E-05	5.264E-05	5.264E-05	5.263E-05	5.260E-05	5.252E-05	5.226E-05	5.150E-05
NB 93M	6.593E-06	1.044E-05	1.406E-05	2.377E-05	3.663E-05	6.293E-05	7.480E-05	7.500E-05	7.496E-05	7.485E-05	7.447E-05	7.339E-05
ZR 95	9.509E 02	1.993E 01	4.176E-01	3.843E-06	1.553E-14	0.0	0.0	0.0	0.0	0.0	0.0	0.0
NB 95	9.162E 02	4.002E 01	8.789E-01	7.870E-06	3.180E-14	0.0	0.0	0.0	0.0	0.0	0.0	0.0
TC 99	1.380E-03	1.385E-03	1.385E-03	1.385E-03	1.385E-03	1.385E-03	1.385E-03	1.384E-03	1.381E-03	1.372E-03	1.341E-03	1.256E-03
RU106	4.630E 00	2.331E 00	1.174E 00	1.498E-01	4.849E-03	5.321E-09	7.362E-30	0.0	0.0	0.0	0.0	0.0
RH106	8.174E 02	3.835E 02	1.931E 02	2.465E 01	7.977E-01	8.753E-07	1.211E-27	0.0	0.0	0.0	0.0	0.0
AG110M	9.267E 00	3.393E 00	1.242E 00	6.096E-02	4.013E-04	7.517E-13	0.0	0.0	0.0	0.0	0.0	0.0
SB125	6.593E 00	5.153E 00	3.998E 00	1.866E 00	5.244E-01	3.272E-03	6.250E-11	0.0	0.0	0.0	0.0	0.0
SN126	8.235E-05	8.235E-05	8.235E-05	8.235E-05	8.234E-05	8.233E-05	8.229E-05	8.218E-05	8.178E-05	8.065E-05	7.683E-05	6.688E-05
SB126	2.285E 00	2.909E-04	2.909E-04	2.909E-04	2.909E-04	2.909E-04	2.907E-04	2.903E-04	2.889E-04	2.849E-04	2.714E-04	2.363E-04
SB126M	7.477E-01	1.372E-03	1.372E-03	1.372E-03	1.372E-03	1.372E-03	1.371E-03	1.369E-03	1.362E-03	1.344E-03	1.280E-03	1.114E-03
I129	2.867E-06	2.902E-06	2.902E-06	2.902E-06	2.902E-06	2.902E-06	2.902E-06	2.902E-06	2.902E-06	2.902E-06	2.901E-06	2.899E-06
CS134	2.101E 02	1.500E 02	1.072E 02	3.906E 01	7.262E 00	9.051E-03	5.343E-13	0.0	0.0	0.0	0.0	0.0
CS135	2.668E-05	2.672E-05	2.672E-05	2.672E-05	2.672E-05	2.672E-05	2.672E-05	2.671E-05	2.671E-05	2.669E-05	2.664E-05	2.648E-05
CS137	1.627E 01	1.590E 01	1.553E 01	1.450E 01	1.292E 01	8.152E 00	1.626E 00	1.625E-02	1.623E-09	1.617E-29	0.0	0.0
BA137M	5.866E 01	5.721E 01	5.591E 01	5.218E 01	4.650E 01	2.934E 01	5.853E 00	5.851E-02	5.841E-09	5.819E-29	0.0	0.0
CE144	1.051E 02	4.316E 01	1.772E 01	1.226E 00	1.431E-02	2.648E-10	0.0	0.0	0.0	0.0	0.0	0.0
PR144	1.221E 03	4.987E 02	2.047E 02	1.417E 01	1.654E-01	3.059E-09	0.0	0.0	0.0	0.0	0.0	0.0
PM147	7.722E 00	6.180E 00	4.745E 00	2.148E 00	5.730E-01	2.907E-03	2.694E-11	0.0	0.0	0.0	0.0	0.0
SM151	6.760E-03	6.808E-03	6.758E-03	6.608E-03	6.366E-03	5.484E-03	3.254E-03	7.322E-04	3.955E-06	1.316E-12	0.0	0.0
EU154	1.330E 01	1.227E 01	1.132E 01	8.886E 00	5.937E 00	1.183E 00	4.129E-03	4.075E-10	0.0	0.0	0.0	0.0
EU155	8.148E-01	7.052E-01	6.103E-01	3.957E-01	1.921E-01	1.068E-02	4.329E-07	1.222E-19	0.0	0.0	0.0	0.0
SUMTOT	5.233E 03	1.324E 03	6.934E 02	2.285E 02	1.359E 02	7.555E 01	1.399E 01	1.256E-01	3.290E-03	3.253E-03	3.139E-03	2.839E-03
TOTAL	2.378E 05	1.327E 03	6.940E 02	2.287E 02	1.359E 02	7.556E 01	1.399E 01	1.256E-01	3.293E-03	3.256E-03	3.140E-03	2.840E-03

Table 2.11. Watts of heavy-element isotopes in a BWR assembly

	DISCHARGE	1. YR	2. YR	5. YR	10. YR	30. YR	100. YR	300. YR	1000. YR	3000. YR	10000. YR	30000. YR
BI214	4.170E-11	7.145E-11	1.106E-10	2.845E-10	7.654E-10	5.166E-09	5.490E-08	5.337E-07	6.133E-06	8.520E-05	2.609E-04	9.120E-04
PO210	4.611E-12	9.259E-12	1.445E-11	4.769E-11	2.242E-10	3.251E-09	7.065E-08	9.801E-07	1.386E-05	1.022E-04	5.898E-04	2.062E-03
PO214	3.501E-10	2.338E-10	3.619E-10	9.311E-10	2.505E-09	1.691E-08	1.797E-07	1.746E-06	2.007E-05	1.479E-04	8.538E-04	2.985E-03
PO218	1.085E-10	1.859E-10	2.877E-10	7.402E-10	1.991E-09	1.344E-08	1.428E-07	1.388E-06	1.595E-05	1.176E-04	6.788E-04	2.373E-03
RN222	9.759E-11	1.672E-10	2.588E-10	6.658E-10	1.791E-09	1.209E-08	1.285E-07	1.249E-06	1.435E-05	1.058E-04	6.106E-04	2.134E-03
RA226	8.471E-11	1.452E-10	2.248E-10	5.784E-10	1.556E-09	1.050E-08	1.116E-07	1.085E-06	1.247E-05	9.189E-05	5.304E-04	1.854E-03
TH230	1.177E-07	1.615E-07	2.055E-07	3.388E-07	5.651E-07	1.520E-06	5.349E-06	1.842E-05	6.826E-05	2.096E-04	6.805E-04	1.839E-03
U234	5.135E-03	5.160E-03	5.185E-03	5.261E-03	5.384E-03	5.829E-03	6.941E-03	8.153E-03	8.467E-03	8.431E-03	8.299E-03	7.938E-03
U236	1.044E-03	1.044E-03	1.044E-03	1.044E-03	1.045E-03	1.046E-03	1.051E-03	1.064E-03	1.108E-03	1.217E-03	1.463E-03	1.667E-03
U238	1.471E-03	1.471E-03	1.471E-03	1.471E-03	1.471E-03	1.471E-03	1.471E-03	1.471E-03	1.471E-03	1.471E-03	1.471E-03	1.471E-03
NP237	1.282E-03	1.307E-03	1.307E-03	1.311E-03	1.322E-03	1.403E-03	1.810E-03	2.829E-03	4.648E-03	5.489E-03	5.512E-03	5.477E-03
PU238	9.554E 00	1.037E 01	1.046E 01	1.026E 01	9.872E 00	8.451E 00	4.906E 00	1.040E 00	4.972E-03	6.314E-08	8.548E-22	0.0
PU239	1.666E 00	1.688E 00	1.688E 00	1.688E 00	1.687E 00	1.687E 00	1.683E 00	1.674E 00	1.643E 00	1.555E 00	1.282E 00	7.314E-01
PU240	2.642E 00	2.642E 00	2.642E 00	2.643E 00	2.644E 00	2.644E 00	2.630E 00	2.577E 00	2.399E 00	1.954E 00	9.532E-01	1.226E-01
PU241	8.461E-01	8.069E-01	7.695E-01	6.673E-01	5.263E-01	2.036E-01	7.338E-03	6.564E-07	9.856E-08	8.334E-08	4.633E-08	8.659E-09
PU242	7.994E-03	7.994E-03	7.994E-03	7.994E-03	7.994E-03	7.994E-03	7.994E-03	7.991E-03	7.982E-03	7.953E-03	7.852E-03	7.570E-03
AM241	7.565E-01	1.818E 00	2.829E 00	5.579E 00	9.342E 00	1.764E 01	2.067E 01	1.516E 01	4.943E 00	2.012E-01	4.000E-05	6.964E-06
AM243	8.205E-02	8.214E-02	8.213E-02	8.211E-02	8.207E-02	8.192E-02	8.140E-02	7.994E-02	7.503E-02	6.259E-02	3.319E-02	5.421E-03
CM242	2.285E 02	4.866E 01	1.031E 01	1.219E-01	2.400E-02	2.186E-02	1.589E-02	6.382E-03	2.622E-04	2.877E-08	3.942E-22	0.0
CM243	7.155E-02	7.002E-02	6.852E-02	6.421E-02	5.762E-02	3.736E-02	8.202E-03	1.077E-04	2.801E-11	4.295E-30	0.0	0.0
CM244	6.630E 00	6.382E 00	6.143E 00	5.476E 00	4.522E 00	2.102E 00	1.440E-01	6.795E-05	1.549E-16	0.0	0.0	0.0
SUMTOT	2.507E 02	7.253E 01	3.501E 01	2.660E 01	2.877E 01	3.288E 01	3.017E 01	2.056E 01	9.088E 00	3.798E 00	2.297E 00	8.977E-01
TOTAL	1.037E 04	7.254E 01	3.502E 01	2.661E 01	2.878E 01	3.289E 01	3.017E 01	2.056E 01	9.092E 00	3.801E 00	2.300E 00	9.030E-01

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Table 2.15. Total curies in BWR assembly as a function of age

Years after discharge	Activation products	Heavy elements	Fission products	Total
0	1.175×10^{5}	5.474×10^{6}	1.998×10^{7}	2.557×10^{7}
1	3.267×10^{3}	2.146×10^{4}	3.149×10^{5}	3.396×10^{5}
2	1.380×10^{3}	1.955×10^{4}	1.729×10^{5}	1.938×10^{5}
5	5.962×10^{2}	1.687×10^{4}	6.924×10^{4}	8.671×10^{4}
10	2.552×10^{2}	1.354×10^{4}	4.666×10^{4}	6.046×10^{4}
30	3.910×10^{1}	5.899×10^{3}	2.673×10^{4}	3.267×10^{4}
100	1.171×10^{1}	1.094×10^{3}	5.044×10^{3}	6.150×10^{3}
300	3.048	6.282×10^{2}	5.414×10^{1}	6.854×10^{2}
1,000	6.367×10^{-1}	2.834×10^{2}	3.337	2.874×10^{2}
3,000	5.671×10^{-1}	1.234×10^{2}	3.284	1.273×10^{2}
10,000	4.408×10^{-1}	7.506×10^{1}	3.218	7.872×10^{1}
30,000	3.236×10^{-1}	2.973×10^{1}	3.039	3.309×10^{1}

Table 2.16. Total watts of thermal power in BWR assembly as a function of age

Years after discharge	Activation products	Heavy elements	Fission products	Total
0	6.080×10^{2}	1.037×10^{4}	2.378×10^{5}	2.488×10^{5}
1	1.430×10^{1}	7.254×10^{1}	1.327×10^{3}	1.414×10^{3}
2	7.459	3.502×10^{1}	6.940×10^{2}	7.365×10^{2}
5	4.500	2.661×10^{1}	2.287×10^{2}	2.598×10^{2}
10	2.189	2.878×10^{1}	1.359×10^{2}	1.669×10^{2}
30	1.675×10^{-1}	3.289×10^{1}	7.556×10^{1}	1.086×10^{2}
100	6.370×10^{-3}	3.017×10^{1}	1.399×10^{1}	4.417×10^{1}
300	2.799×10^{-3}	2.056×10^{1}	1.256×10^{-1}	2.069×10^{1}
1,000	1.802×10^{-3}	9.092	3.293×10^{-3}	9.097
3,000	1.693×10^{-3}	3.801	3.254×10^{-3}	3.806
10,000	1.420×10^{-3}	2.300	3.140×10^{-3}	2.305
30,000	9.968×10^{-4}	9.030×10^{-1}	2.840×10^{-3}	9.068×10^{-1}

Table 2.17. Spontaneous fission neutron source in a BWR assembly (neutrons/sec)

	DISCHARGE	1. YR	2. YR	5. YR	10. YR	30. YR	100. YR	300. YR	1000. YR	3000. YR	10000. YR	30000. YR
PU238	4.282E 04	4.646E 04	4.687E 04	4.599E 04	4.424E 04	3.787E 04	2.199E 04	4.661E 03	2.228E 01	2.830E-04	3.831E-18	0.0
PU240	3.327E 05	3.327E 05	3.327E 05	3.328E 05	3.330E 05	3.330E 05	3.312E 05	3.245E 05	3.020E 05	2.460E 05	1.200E 05	1.544E 04
PU242	1.150E 05	1.150E 05	1.150E 05	1.150E 05	1.150E 05	1.150E 05	1.150E 05	1.149E 05	1.148E 05	1.144E 05	1.129E 05	1.089E 05
CM242	3.670E 07	7.816E 06	1.657E 06	1.958E 04	3.856E 03	3.512E 03	2.552E 03	1.025E 03	4.212E 01	4.622E-03	6.333E-17	0.0
CM244	2.592E 07	2.495E 07	2.402E 07	2.141E 07	1.768E 07	8.219E 06	5.631E 05	2.657E 02	6.057E-10	0.0	0.0	0.0
CM246	8.993E 04	8.992E 04	8.991E 04	8.987E 04	8.980E 04	8.954E 04	8.862E 04	8.605E 04	7.763E 04	5.788E 04	2.065E 04	1.089E 03
TOTALS												
TABLE	6.320E 07	3.335E 07	2.626E 07	2.201E 07	1.826E 07	8.798E 06	1.123E 06	5.315E 05	4.946E 05	4.183E 05	2.537E 05	1.255E 05
ACTUAL	6.320E 07	3.335E 07	2.626E 07	2.201E 07	1.826E 07	8.798E 06	1.123E 06	5.315E 05	4.946E 05	4.183E 05	2.537E 05	1.255E 05

Table 2.18. Alpha, n neutron source in a BWR assembly (neutrons/sec)

	DISCHARGE	1. YR	2. YR	5. YR	10. YR	30. YR	100. YR	300. YR	1000. YR	3000. YR	10000. YR	30000. YR
PO210	2.401E-07	4.822E-07	7.527E-07	2.883E-06	1.167E-05	1.693E-04	3.679E-03	5.104E-02	7.219E-01	5.321E 00	3.071E 01	1.074E 02
PO213	2.715E-04	9.049E-05	9.388E-05	1.052E-04	1.272E-04	2.540E-04	1.247E-03	1.047E-02	1.624E-01	1.980E 00	2.187E 01	1.268E 02
PO214	4.859E-05	3.246E-05	5.023E-05	1.292E-04	3.476E-04	2.347E-03	2.494E-02	2.428E-01	2.786E 00	2.053E 01	1.185E 02	4.143E 02
PO218	8.199E-06	1.405E-05	2.174E-05	5.593E-05	1.505E-04	1.016E-03	1.079E-02	1.049E-01	1.206E 00	8.887E 00	5.129E 01	1.793E 02
AT217	1.495E-04	4.981E-05	5.168E-05	5.791E-05	7.000E-05	1.399E-04	6.866E-04	5.762E-03	8.941E-02	1.090E 00	1.204E 01	6.983E 01
RN222	5.570E-06	9.544E-06	1.477E-05	3.800E-05	1.022E-04	6.900E-04	7.333E-03	7.129E-02	8.192E-01	6.038E 00	3.485E 01	1.218E 02
FR221	9.670E-05	3.223E-05	3.344E-05	3.747E-05	4.529E-05	9.049E-05	4.442E-04	3.728E-03	5.785E-02	7.053E-01	7.788E 00	4.518E 01
RA226	3.329E-06	5.709E-06	8.835E-06	2.273E-05	6.115E-05	4.127E-04	4.386E-03	4.264E-02	4.900E-01	3.611E 00	2.085E 01	7.287E 01
AC225	7.214E-05	2.404E-05	2.495E-05	2.795E-05	3.379E-05	6.751E-05	3.314E-04	2.781E-03	4.316E-02	5.262E-01	5.810E 00	3.371E 01
TH230	4.607E-03	6.320E-03	8.041E-03	1.326E-02	2.211E-02	5.946E-02	2.093E-01	7.209E-01	2.671E 00	8.201E 00	2.662E 01	7.196E 01
U234	2.110E 02	2.120E 02	2.131E 02	2.162E 02	2.212E 02	2.395E 02	2.852E 02	3.350E 02	3.479E 02	3.465E 02	3.411E 02	3.262E 02
U236	3.659E 01	3.660E 01	3.660E 01	3.661E 01	3.662E 01	3.666E 01	3.683E 01	3.729E 01	3.883E 01	4.266E 01	5.129E 01	5.842E 01
U238	4.292E 01	4.292E 01	4.292E 01	4.292E 01	4.292E 01	4.292E 01	4.292E 01	4.292E 01	4.292E 01	4.292E 01	4.292E 01	4.292E 01
NP237	5.563E 01	5.667E 01	5.670E 01	5.686E 01	5.733E 01	6.084E 01	7.853E 01	1.227E 02	2.016E 02	2.381E 02	2.391E 02	2.376E 02
PU238	4.221E 05	4.580E 05	4.621E 05	4.534E 05	4.362E 05	3.738E 05	2.168E 05	4.595E 04	2.197E 02	2.790E-03	3.777E-17	0.0
PU239	5.889E 04	5.965E 04	5.965E 04	5.965E 04	5.964E 04	5.961E 04	5.949E 04	5.917E 04	5.805E 04	5.496E 04	4.530E 04	2.585E 04
PU240	9.073E 04	9.074E 04	9.075E 04	9.078E 04	9.081E 04	9.082E 04	9.033E 04	8.851E 04	8.238E 04	6.711E 04	3.274E 04	8.211E 03
PU241	5.053E 04	4.818E 04	4.595E 04	3.985E 04	3.143E 04	1.216E 04	4.382E 02	3.920E-02	5.886E-03	4.977E-03	2.767E-03	5.171E-04
PU242	2.695E 02	2.695E 02	2.695E 02	2.695E 02	2.695E 02	2.695E 02	2.694E 02	2.694E 02	2.690E 02	2.681E 02	2.647E 02	2.552E 02
AM241	4.601E 04	1.106E 05	1.720E 05	3.393E 05	5.681E 05	1.073E 06	1.257E 06	9.218E 05	3.006E 05	1.224E 04	2.433E 00	4.235E-01
AM243	6.327E 03	6.334E 03	6.333E 03	6.332E 03	6.329E 03	6.317E 03	6.277E 03	6.165E 03	5.786E 03	4.827E 03	2.560E 03	4.180E 02
CM242	3.744E 07	7.974E 06	1.690E 06	1.998E 04	3.933E 03	3.583E 03	2.604E 03	1.046E 03	4.297E 01	4.715E-03	6.461E-17	0.0
CM243	5.621E 03	5.500E 03	5.382E 03	5.044E 03	4.526E 03	2.935E 03	6.443E 02	8.462E 00	2.200E-06	3.374E-25	0.0	0.0
CM244	1.007E 06	9.694E 05	9.330E 05	8.317E 05	6.868E 05	3.193E 05	2.188E 04	1.032E 01	2.353E-11	0.0	0.0	0.0
TOTALS												
TABLE	3.913E 07	9.723E 06	3.466E 06	1.847E 06	1.888E 06	1.941E 06	1.656E 06	1.123E 06	4.880E 05	1.401E 05	8.190E 04	3.273E 04
ACTUAL	3.913E 07	9.723E 06	3.466E 06	1.847E 06	1.888E 06	1.941E 06	1.656E 06	1.123E 06	4.880E 05	1.401E 05	8.190E 04	3.273E 04

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Table 2.19. Total grams charged to a PWR fuel assembly[a]

Elements or isotopes	Grams charged
C	7.353×10^{1}
N	5.302×10^{1}
O	6.214×10^{4}
Si	2.107×10^{2}
P	1.485×10^{2}
Cr	5.081×10^{3}
Mn	4.003×10^{2}
Fe	1.460×10^{4}
Ni	5.740×10^{3}
Zr	1.062×10^{5}
Nb	3.280×10^{2}
Mo	1.818×10^{2}
Sn	1.737×10^{3}
^{235}U	1.476×10^{4}
^{238}U	4.465×10^{5}

[a]Impurities of 25 ppm of nitrogen and 1 ppm of lithium, in addition to others, are found in the fuel.

Table 2.20. Grams of fission-product elements in a PWR assembly

	DISCHARGE	1. YR	2. YR	5. YR	10. YR	30. YR	100. YR	300. YR	1000. YR	3000. YR	10000. YR	30000. YR
H	2.639E-02	2.498E-02	2.357E-02	1.991E-02	1.502E-02	4.866E-03	2.418E-05	1.200E-09	8.859E-27	0.0	0.0	0.0
LI	8.934E-05	8.934E-05	8.934E-05	8.934E-05	8.934E-05	8.934E-05	8.934E-05	8.934E-05	8.934E-05	8.934E-05	8.934E-05	8.934E-05
BE	6.895E-05	6.895E-05	6.895E-05	6.895E-05	6.895E-05	6.895E-05	6.895E-05	6.894E-05	6.893E-05	6.890E-05	6.878E-05	6.845E-05
C	1.212E-05	1.212E-05	1.212E-05	1.212E-05	1.211E-05	1.208E-05	1.198E-05	1.169E-05	1.074E-05	8.433E-06	3.616E-06	3.218E-07
CO	7.299E-13	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
NI	2.619E-10	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
CU	2.329E-09	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
ZN	3.273E-05	1.743E-08	1.743E-08	1.743E-08	1.743E-08	1.743E-08	1.743E-08	1.743E-08	1.743E-08	1.743E-08	1.743E-08	1.743E-08
GA	1.794E-05	4.093E-07	4.093E-07	4.093E-07	4.093E-07	4.093E-07	4.093E-07	4.093E-07	4.093E-07	4.093E-07	4.093E-07	4.093E-07
GE	3.047E-01	3.045E-01	3.045E-01	3.045E-01	3.045E-01	3.045E-01	3.045E-01	3.045E-01	3.045E-01	3.045E-01	3.045E-01	3.045E-01
AS	9.348E-02	9.229E-02	9.229E-02	9.229E-02	9.229E-02	9.229E-02	9.229E-02	9.229E-02	9.229E-02	9.229E-02	9.229E-02	9.229E-02
SE	2.598E 01	2.598E 01	2.598E 01	2.598E 01	2.598E 01	2.598E 01	2.598E 01	2.597E 01	2.595E 01	2.589E 01	2.570E 01	2.524E 01
BR	9.990E 00	9.984E 00	9.984E 00	9.984E 00	9.984E 00	9.985E 00	9.987E 00	9.992E 00	1.001E 01	1.007E 01	1.026E 01	1.073E 01
KR	1.703E 02	1.696E 02	1.689E 02	1.672E 02	1.650E 02	1.608E 02	1.592E 02	1.591E 02	1.591E 02	1.591E 02	1.591E 02	1.591E 02
RB	1.572E 02	1.579E 02	1.586E 02	1.603E 02	1.625E 02	1.668E 02	1.684E 02	1.684E 02	1.684E 02	1.684E 02	1.684E 02	1.684E 02
SR	4.219E 02	4.029E 02	3.970E 02	3.802E 02	3.589E 02	2.797E 02	1.828E 02	1.620E 02	1.619E 02	1.619E 02	1.619E 02	1.619E 02
Y	2.174E 02	2.107E 02	2.105E 02	2.105E 02	2.105E 02	2.105E 02	2.105E 02	2.105E 02	2.105E 02	2.105E 02	2.105E 02	2.105E 02
ZR	1.631E 03	1.624E 03	1.630E 03	1.647E 03	1.672E 03	1.747E 03	1.844E 03	1.865E 03	1.865E 03	1.864E 03	1.863E 03	1.858E 03
NB	1.797E 01	7.815E-01	1.830E-02	1.886E-03	3.096E-03	7.936E-03	2.487E-02	7.326E-02	2.426E-01	7.258E-01	2.412E 00	7.181E 00
MO	1.485E 03	1.533E 03	1.535E 03	1.535E 03	1.535E 03	1.535E 03	1.535E 03	1.535E 03	1.535E 03	1.535E 03	1.535E 03	1.535E 03
TC	3.578E 02	3.594E 02	3.594E 02	3.594E 02	3.594E 02	3.594E 02	3.593E 02	3.590E 02	3.582E 02	3.559E 02	3.478E 02	3.258E 02
RU	1.104E 03	1.044E 03	1.024E 03	1.007E 03	1.005E 03	1.005E 03	1.005E 03	1.005E 03	1.006E 03	1.008E 03	1.016E 03	1.038E 03
RH	1.966E 02	2.175E 02	2.175E 02	2.175E 02	2.175E 02	2.175E 02	2.175E 02	2.175E 02	2.175E 02	2.175E 02	2.175E 02	2.175E 02
PD	5.525E 02	5.916E 02	6.110E 02	6.282E 02	6.307E 02	6.308E 02	6.308E 02	6.308E 02	6.307E 02	6.307E 02	6.306E 02	6.304E 02
AG	3.584E 01	3.543E 01	3.534E 01	3.528E 01	3.528E 01	3.528E 01	3.528E 01	3.528E 01	3.529E 01	3.531E 01	3.539E 01	3.560E 01
CD	4.887E 01	4.929E 01	4.938E 01	4.942E 01	4.941E 01	4.936E 01	4.933E 01	4.933E 01	4.933E 01	4.933E 01	4.933E 01	4.933E 01
IN	1.056E 00	1.124E 00	1.129E 00	1.144E 00	1.164E 00	1.210E 00	1.237E 00	1.238E 00	1.238E 00	1.238E 00	1.238E 00	1.238E 00
SN	4.182E 01	4.131E 01	4.125E 01	4.124E 01	4.123E 01	4.123E 01	4.123E 01	4.121E 01	4.115E 01	4.098E 01	4.039E 01	3.887E 01
SB	1.465E 01	1.384E 01	1.236E 01	1.027E 01	8.987E 00	8.438E 00	8.431E 00	8.432E 00	8.432E 00	8.432E 00	8.432E 00	8.432E 00
TE	2.210E 02	2.185E 02	2.195E 02	2.216E 02	2.229E 02	2.234E 02	2.235E 02	2.235E 02	2.235E 02	2.237E 02	2.243E 02	2.258E 02
I	1.106E 02	1.082E 02	1.083E 02	1.083E 02	1.083E 02	1.083E 02	1.083E 02	1.083E 02	1.083E 02	1.083E 02	1.083E 02	1.082E 02
XE	2.446E 03	2.446E 03	2.446E 03	2.446E 03	2.446E 03	2.446E 03	2.446E 03	2.446E 03	2.446E 03	2.446E 03	2.446E 03	2.447E 03
CS	1.269E 03	1.246E 03	1.223E 03	1.170E 03	1.108E 03	9.445E 02	7.231E 02	6.685E 02	6.679E 02	6.678E 02	6.675E 02	6.666E 02
BA	6.387E 02	6.574E 02	6.809E 02	7.339E 02	7.957E 02	9.598E 02	1.181E 03	1.235E 03	1.236E 03	1.236E 03	1.236E 03	1.237E 03
LA	5.628E 02	5.613E 02	5.613E 02	5.613E 02	5.613E 02	5.613E 02	5.613E 02	5.613E 02	5.613E 02	5.613E 02	5.613E 02	5.613E 02
CE	1.285E 03	1.168E 03	1.126E 03	1.098E 03	1.096E 03	1.096E 03	1.096E 03	1.096E 03	1.096E 03	1.096E 03	1.096E 03	1.096E 03
PR	5.003E 02	5.154E 02	5.154E 02	5.154E 02	5.154E 02	5.154E 02	5.154E 02	5.154E 02	5.154E 02	5.154E 02	5.154E 02	5.154E 02
ND	1.675E 03	1.784E 03	1.826E 03	1.854E 03	1.856E 03	1.856E 03	1.856E 03	1.856E 03	1.856E 03	1.856E 03	1.856E 03	1.856E 03
PM	6.663E 01	5.215E 01	4.004E 01	1.812E 01	4.835E 00	2.458E-02	2.278E-10	0.0	0.0	0.0	0.0	0.0
SM	3.350E 02	3.526E 02	3.647E 02	3.864E 02	3.995E 02	4.030E 02	4.012E 02	3.986E 02	3.979E 02	3.979E 02	3.979E 02	3.979E 02
EU	7.588E 01	7.258E 01	7.055E 01	6.563E 01	6.018E 01	5.310E 01	5.365E 01	5.619E 01	5.692E 01	5.692E 01	5.692E 01	5.692E 01
GD	3.623E 01	4.006E 01	4.213E 01	4.720E 01	5.289E 01	6.086E 01	6.256E 01	6.256E 01	6.256E 01	6.256E 01	6.256E 01	6.256E 01
TB	1.214E 00	1.174E 00	1.172E 00	1.172E 00	1.172E 00	1.172E 00	1.172E 00	1.172E 00	1.172E 00	1.172E 00	1.172E 00	1.172E 00
DY	5.900E-01	6.328E-01	6.339E-01	6.340E-01	6.340E-01	6.340E-01	6.340E-01	6.340E-01	6.340E-01	6.340E-01	6.340E-01	6.340E-01
HO	6.303E-02	6.296E-02	6.296E-02	6.296E-02	6.296E-02	6.295E-02	6.293E-02	6.286E-02	6.269E-02	6.244E-02	6.233E-02	6.233E-02
ER	2.518E-02	2.534E-02	2.534E-02	2.534E-02	2.534E-02	2.535E-02	2.537E-02	2.544E-02	2.561E-02	2.585E-02	2.596E-02	2.597E-02
TM	2.747E-05	2.574E-05	2.522E-05	2.504E-05	2.499E-05	2.499E-05	2.499E-05	2.499E-05	2.499E-05	2.499E-05	2.499E-05	2.499E-05
YB	3.972E-06	7.210E-06	7.728E-06	7.909E-06	7.955E-06	7.964E-06	7.964E-06	7.964E-06	7.964E-06	7.964E-06	7.964E-06	7.964E-06
TOTAL	1.571E 04	1.571E 04	1.571E 04	1.571E 04	1.571E 04	1.571E 04	1.571E 04	1.571E 04	1.571E 04	1.571E 04	1.571E 04	1.571E 04

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Table 2.21. Grams of heavy elements in a PWR assembly

	DISCHARGE	1. YR	2. YR	5. YR	10. YR	30. YR	100. YR	300. YR	1000. YR	3000. YR	10000. YR	30000. YR
HE	1.041E-01	1.903E-01	2.228E-01	2.828E-01	3.825E-01	8.231E-01	2.364E 00	5.674E 00	1.194E 01	1.910E 01	3.283E 01	5.210E 01
SP	5.654E-07	1.294E-06	1.796E-06	3.051E-06	4.833E-06	9.424E-06	1.367E-05	1.583E-05	2.210E-05	3.808E-05	7.834E-05	1.394E-04
TL	5.648E-13	1.633E-12	2.818E-12	5.989E-12	8.775E-12	8.739E-12	4.542E-12	9.207E-13	9.174E-13	2.695E-12	9.021E-12	2.623E-11
PB	1.048E-07	4.550E-07	1.178E-06	5.570E-06	1.802E-05	7.843E-05	2.240E-04	3.600E-04	5.966E-04	5.057E-03	1.006E-01	1.168E 00
BI	4.906E-11	1.192E-10	1.936E-10	3.976E-10	6.070E-10	9.784E-10	6.551E-09	1.451E-07	6.981E-06	2.583E-04	1.045E-02	2.088E-01
PO	2.873E-14	7.843E-14	1.589E-13	7.339E-13	3.993E-12	6.339E-11	1.440E-09	2.054E-08	2.964E-07	2.199E-06	1.272E-05	4.443E-05
AT	4.891E-20	1.340E-20	1.381E-20	1.531E-20	1.853E-20	4.057E-20	2.443E-19	2.251E-18	3.503E-17	4.210E-16	4.615E-15	2.672E-14
RN	5.370E-13	1.514E-12	2.608E-12	5.624E-12	8.698E-12	1.446E-11	7.829E-11	7.418E-10	8.674E-09	6.436E-08	3.722E-07	1.300E-06
FR	5.091E-16	2.615E-16	3.428E-16	5.859E-16	9.885E-16	2.596E-15	8.922E-15	3.955E-14	3.846E-13	4.044E-12	4.290E-11	2.466E-10
RA	8.790E-09	1.941E-08	3.259E-08	8.198E-08	1.917E-07	1.097E-06	1.160E-05	1.154E-04	1.351E-03	1.002E-02	5.796E-02	2.025E-01
AC	2.283E-09	5.209E-09	8.117E-09	1.673E-08	3.074E-08	8.353E-08	2.509E-07	7.101E-07	2.384E-06	6.945E-06	2.261E-05	6.309E-05
TH	5.845E-04	8.661E-04	1.150E-03	2.008E-03	3.463E-03	9.571E-03	3.394E-02	1.167E-01	4.325E-01	1.339E 00	4.460E 00	1.263E 01
PA	1.501E-04	1.542E-04	1.578E-04	1.683E-04	1.859E-04	2.565E-04	5.042E-04	1.211E-03	3.671E-03	1.064E-02	3.444E-02	9.546E-02
U	4.412E 05	4.412E 05	4.412E 05	4.412E 05	4.412E 05	4.412E 05	4.412E 05	4.413E 05	4.414E 05	4.417E 05	4.425E 05	4.436E 05
NP	2.392E 02	2.036E 02	2.037E 02	2.041E 02	2.056E 02	2.165E 02	2.714E 02	4.088E 02	6.541E 02	7.676E 02	7.707E 02	7.657E 02
PU	4.187E 03	4.206E 03	4.181E 03	4.112E 03	4.016E 03	3.792E 03	3.630E 03	3.568E 03	3.447E 03	3.146E 03	2.364E 03	1.262E 03
AM	5.449E 01	8.062E 01	1.055E 02	1.731E 02	2.657E 02	4.697E 02	5.444E 02	4.090E 02	1.570E 02	3.514E 01	1.603E 01	2.618E 00
CM	1.425E 01	9.776E 00	8.574E 00	7.451E 00	6.176E 00	2.944E 00	3.052E-01	9.835E-02	9.073E-02	7.309E-02	3.533E-02	5.461E-03
BK	4.611E-08	2.092E-08	9.482E-09	8.833E-10	1.691E-11	2.271E-18	6.322E-23	6.271E-23	6.099E-23	5.632E-23	4.261E-23	1.921E-23
CF	1.691E-08	4.131E-08	5.195E-08	5.852E-08	5.683E-08	5.092E-08	4.268E-08	2.893E-08	7.561E-09	2.405E-10	4.662E-13	7.789E-19
ES	2.561E-12	2.328E-14	9.254E-15	5.887E-16	5.968E-18	6.305E-26	0.0	0.0	0.0	0.0	0.0	0.0
TOTAL	4.457E 05	4.457E 05	4.457E 05	4.457E 05	4.457E 05	4.457E 05	4.457E 05	4.457E 05	4.457E 05	4.457E 05	4.457E 05	4.457E 05

Table 2.22. Curies of light-element isotopes in a PWR assembly

	DISCHARGE	1. YR	2. YR	5. YR	10. YR	30. YR	100. YR	300. YR	1000. YR	3000. YR	10000. YR	30000. YR
H 3	1.214E 02	1.148E 02	1.085E 02	9.161E 01	6.911E 01	2.239E 01	4.334E-01	5.522E-06	4.077E-23	0.0	0.0	0.0
C 14	6.853E-01	6.852E-01	6.851E-01	6.848E-01	6.844E-01	6.828E-01	6.770E-01	6.608E-01	6.072E-01	4.767E-01	2.044E-01	1.819E-02
CL 36	5.126E-03	5.126E-03	5.126E-03	5.126E-03	5.126E-03	5.126E-03	5.125E-03	5.123E-03	5.115E-03	5.092E-03	5.013E-03	4.794E-03
MN 54	4.532E 01	1.965E 01	8.522E 00	6.948E-01	1.066E-02	5.889E-10	0.0	0.0	0.0	0.0	0.0	0.0
FE 55	2.572E 03	1.970E 03	1.509E 03	6.782E 02	1.788E 02	8.644E-01	6.793E-09	0.0	0.0	0.0	0.0	0.0
CO 58	3.157E 03	9.062E 01	2.601E 00	6.150E-05	1.198E-12	0.0	0.0	0.0	0.0	0.0	0.0	0.0
CO 60	3.629E 03	3.181E 03	2.788E 03	1.878E 03	9.716E 02	6.964E 01	6.867E-03	2.459E-18	0.0	0.0	0.0	0.0
NI 59	2.118E 00	2.118E 00	2.118E 00	2.117E 00	2.117E 00	2.117E 00	2.116E 00	2.112E 00	2.099E 00	2.063E 00	1.942E 00	1.633E 00
NI 63	3.061E 02	3.038E 02	3.015E 02	2.948E 02	2.839E 02	2.442E 02	1.441E 02	3.193E 01	1.637E-01	4.677E-08	0.0	0.0
ZN 65	6.348E 01	2.259E 01	8.037E 00	3.620E-01	2.068E-03	2.187E-12	0.0	0.0	0.0	0.0	0.0	0.0
ZR 93	1.049E-01	1.049E-01	1.049E-01	1.049E-01	1.049E-01	1.049E-01	1.049E-01	1.049E-01	1.049E-01	1.047E-01	1.042E-01	1.027E-01
ZR 95	2.142E 04	4.489E 02	9.406E 00	8.656E-05	3.498E-13	0.0	0.0	0.0	0.0	0.0	0.0	0.0
NB 93M	6.576E-03	1.181E-02	1.674E-02	2.995E-02	4.766E-02	8.325E-02	9.940E-02	9.967E-02	9.962E-02	9.948E-02	9.897E-02	9.753E-02
NB 94	6.242E-01	6.242E-01	6.242E-01	6.241E-01	6.240E-01	6.236E-01	6.220E-01	6.177E-01	6.029E-01	5.625E-01	4.413E-01	2.205E-01
NB 95	2.190E 04	9.494E 02	2.085E 01	1.867E-04	7.543E-13	0.0	0.0	0.0	0.0	0.0	0.0	0.0
MO 93	1.396E-02	1.396E-02	1.396E-02	1.395E-02	1.393E-02	1.387E-02	1.364E-02	1.303E-02	1.108E-02	6.981E-03	1.385E-03	1.362E-05
TC 99	2.575E-03	2.589E-03	2.589E-03	2.589E-03	2.589E-03	2.588E-03	2.588E-03	2.586E-03	2.580E-03	2.563E-03	2.505E-03	2.347E-03
IN113M	4.128E 02	4.568E 01	5.054E 00	6.844E-03	1.105E-07	8.341E-27	0.0	0.0	0.0	0.0	0.0	0.0
SN113	4.126E 02	4.565E 01	5.051E 00	6.659E-03	1.104E-07	8.336E-27	0.0	0.0	0.0	0.0	0.0	0.0
SN119M	3.082E 03	1.097E 03	3.902E 02	1.758E 01	1.005E-01	1.066E-10	0.0	0.0	0.0	0.0	0.0	0.0
SN123	1.489E 02	2.094E 01	2.944E 00	8.160E-03	4.484E-07	4.090E-24	0.0	0.0	0.0	0.0	0.0	0.0
SB125	7.875E 02	6.187E 02	4.800E 02	2.241E 02	6.296E 01	3.927E-01	7.501E-09	0.0	0.0	0.0	0.0	0.0
TE125M	1.671E 02	1.507E 02	1.172E 02	5.472E 01	1.538E 01	9.589E-02	1.832E-09	0.0	0.0	0.0	0.0	0.0
SUMTOT	5.823E 04	9.082E 03	5.761E 03	3.243E 03	1.585E 03	3.412E 02	1.482E 02	3.555E 01	3.696E 00	3.321E 00	2.800E 00	2.079E 00
TOTAL	2.070E 05	9.098E 03	5.762E 03	3.244E 03	1.586E 03	3.414E 02	1.482E 02	3.555E 01	3.696E 00	3.321E 00	2.800E 00	2.079E 00

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Table 2.23. Curies of fission-product isotopes in a PWR assembly

	DISCHARGE	1. YR	2. YR	5. YR	10. YR	30. YR	100. YR	300. YR	1000. YR	3000. YR	10000. YR	30000. YR
H 3	2.557E 02	2.417E 02	2.285E 02	1.929E 02	1.456E 02	4.716E 01	9.129E-01	1.163E-05	8.586E-23	0.0	0.0	0.0
SE 79	1.888E-01	1.888E-01	1.888E-01	1.888E-01	1.888E-01	1.887E-01	1.886E-01	1.882E-01	1.868E-01	1.828E-01	1.697E-01	1.371E-01
KR 85	4.367E 03	4.095E 03	3.839E 03	3.162E 03	2.289E 03	6.289E 02	6.834E 00	1.673E-05	3.839E-25	0.0	0.0	0.0
SR 89	3.641E 05	2.799E 03	2.149E 01	9.756E-06	2.615E-16	0.0	0.0	0.0	0.0	0.0	0.0	0.0
SR 90	3.493E 04	3.408E 04	3.325E 04	3.087E 04	2.729E 04	1.666E 04	2.964E 03	2.138E 01	6.780E-07	2.549E-28	0.0	0.0
Y 90	3.648E 04	3.409E 04	3.325E 04	3.088E 04	2.730E 04	1.667E 04	2.965E 03	2.138E 01	6.782E-07	2.550E-28	0.0	0.0
Y 91	4.784E 05	6.403E 03	8.511E 01	2.000E-04	8.308E-14	0.0	0.0	0.0	0.0	0.0	0.0	0.0
ZR 93	1.342E 00	1.343E 00	1.343E 00	1.343E 00	1.343E 00	1.343E 00	1.342E 00	1.342E 00	1.342E 00	1.340E 00	1.333E 00	1.313E 00
NB 93M	9.427E-02	1.606E-01	2.232E-01	3.908E-01	6.128E-01	1.067E 00	1.272E 00	1.275E 00	1.275E 00	1.273E 00	1.266E 00	1.248E 00
ZR 95	6.893E 05	1.445E 04	3.027E 02	2.786E-03	1.126E-11	0.0	0.0	0.0	0.0	0.0	0.0	0.0
NB 95	7.013E 05	3.056E 04	6.710E 02	6.007E-03	2.428E-11	0.0	0.0	0.0	0.0	0.0	0.0	0.0
TC 99	6.095E 00	6.124E 00	6.124E 00	6.124E 00	6.124E 00	6.124E 00	6.122E 00	6.118E 00	6.104E 00	6.065E 00	5.927E 00	5.552E 00
RU103	6.872E 05	1.149E 03	1.955E 00	9.130E-09	1.191E-22	0.0	0.0	0.0	0.0	0.0	0.0	0.0
RH103M	6.874E 05	1.150E 03	1.957E 00	9.139E-09	1.192E-22	0.0	0.0	0.0	0.0	0.0	0.0	0.0
RU106	2.597E 05	1.308E 05	6.584E 04	8.405E 03	2.720E 02	2.985E-04	4.130E-25	0.0	0.0	0.0	0.0	0.0
RH106	2.867E 05	1.308E 05	6.584E 04	8.405E 03	2.720E 02	2.985E-04	4.130E-25	0.0	0.0	0.0	0.0	0.0
PD107	5.180E-02	5.181E-02	5.181E-02	5.181E-02	5.181E-02	5.181E-02	5.181E-02	5.181E-02	5.180E-02	5.179E-02	5.175E-02	5.164E-02
SB125	6.802E 03	5.323E 03	4.129E 03	1.928E 03	5.416E 02	3.374E 00	6.445E-08	0.0	0.0	0.0	0.0	0.0
TE125M	1.439E 03	1.297E 03	1.008E 03	4.708E 02	1.323E 02	8.242E-01	1.574E-08	0.0	0.0	0.0	0.0	0.0
SN126	3.577E-01	3.577E-01	3.577E-01	3.577E-01	3.577E-01	3.576E-01	3.575E-01	3.570E-01	3.553E-01	3.504E-01	3.338E-01	2.905E-01
SB126	5.067E 02	5.008E-02	5.008E-02	5.008E-02	5.008E-02	5.007E-02	5.005E-02	4.998E-02	4.973E-02	4.905E-02	4.673E-02	4.067E-02
SB126M	2.283E 02	3.577E-01	3.577E-01	3.577E-01	3.577E-01	3.576E-01	3.575E-01	3.570E-01	3.552E-01	3.504E-01	3.338E-01	2.905E-01
I129	1.422E-02	1.444E-02	1.444E-02	1.444E-02	1.444E-02	1.444E-02	1.444E-02	1.444E-02	1.444E-02	1.443E-02	1.443E-02	1.442E-02
CS134	7.127E 04	5.091E 04	3.637E 04	1.325E 04	2.464E 03	3.003E 00	1.773E-10	0.0	0.0	0.0	0.0	0.0
CS135	1.711E-01	1.714E-01	1.714E-01	1.714E-01	1.714E-01	1.714E-01	1.714E-01	1.714E-01	1.714E-01	1.713E-01	1.709E-01	1.699E-01
CS137	4.786E 04	4.677E 04	4.570E 04	4.265E 04	3.801E 04	2.398E 04	4.785E 03	4.783E 01	4.777E-06	4.759E-26	0.0	0.0
BA137M	4.537E 04	4.424E 04	4.324E 04	4.035E 04	3.596E 04	2.269E 04	4.526E 03	4.525E 01	4.519E-06	4.502E-26	0.0	0.0
CE144	5.577E 05	2.290E 05	9.400E 04	6.504E 03	7.587E 01	1.404E-06	0.0	0.0	0.0	0.0	0.0	0.0
PR144	5.618E 05	2.290E 05	9.400E 04	6.504E 03	7.587E 01	1.404E-06	0.0	0.0	0.0	0.0	0.0	0.0
PR144M	6.699E 03	2.748E 03	1.128E 03	7.805E 01	9.104E-01	1.684E-08	0.0	0.0	0.0	0.0	0.0	0.0
PM147	5.980E 04	4.837E 04	3.714E 04	1.681E 04	4.485E 03	2.280E 01	2.112E-07	0.0	0.0	0.0	0.0	0.0
SM151	1.731E 02	1.749E 02	1.736E 02	1.697E 02	1.635E 02	1.409E 02	8.358E 01	1.881E 01	1.017E-01	3.383E-08	0.0	0.0
EU154	4.902E 03	4.522E 03	4.172E 03	3.275E 03	2.188E 03	4.360E 02	1.533E 00	1.513E-07	0.0	0.0	0.0	0.0
EU155	3.205E 03	2.774E 03	2.401E 03	1.556E 03	7.557E 02	4.202E 01	1.703E-03	4.808E-16	0.0	0.0	0.0	0.0
SUMTOT	5.598E 06	1.056E 06	5.669E 05	2.155E 05	1.424E 05	6.134E 04	1.534E 04	1.646E 02	1.001E 01	9.847E 00	9.647E 00	9.108E 00
TOTAL	7.293E 07	1.059E 06	5.673E 05	2.155E 05	1.425E 05	6.135E 04	1.534E 04	1.646E 02	1.001E 01	9.847E 00	9.647E 00	9.108E 00

Table 2.24. Curies of heavy-element isotopes in a PWR assembly

	DISCHARGE	1. YR	2. YR	5. YR	10. YR	30. YR	100. YR	300. YR	1000. YR	3000. YR	10000. YR	30000. YR
PB210	2.206E-10	4.882E-10	9.396E-10	3.996E-09	1.790E-08	2.856E-07	6.489E-06	9.259E-05	1.336E-03	9.914E-03	5.733E-02	2.003E-01
PB214	5.873E-09	1.104E-08	1.814E-08	5.113E-08	1.457E-07	1.041E-06	1.145E-05	1.142E-04	1.336E-03	9.914E-03	5.733E-02	2.003E-01
BI210	2.227E-10	4.885E-10	9.400E-10	3.997E-09	1.791E-08	2.856E-07	6.489E-06	9.259E-05	1.336E-03	9.914E-03	5.733E-02	2.003E-01
BI214	5.873E-09	1.104E-08	1.814E-08	5.113E-08	1.457E-07	1.041E-06	1.145E-05	1.142E-04	1.336E-03	9.914E-03	5.733E-02	2.003E-01
PO210	1.235E-10	3.363E-10	6.862E-10	3.245E-09	1.791E-08	2.856E-07	6.489E-06	9.259E-05	1.336E-03	9.914E-03	5.733E-02	2.003E-01
PO214	8.320E-09	1.104E-08	1.814E-08	5.113E-08	1.457E-07	1.041E-06	1.145E-05	1.142E-04	1.336E-03	9.914E-03	5.733E-02	2.003E-01
PO218	5.873E-09	1.104E-08	1.814E-08	5.113E-08	1.457E-07	1.041E-06	1.145E-05	1.142E-04	1.336E-03	9.914E-03	5.733E-02	2.003E-01
RN222	5.873E-09	1.104E-08	1.814E-08	5.113E-08	1.457E-07	1.041E-06	1.145E-05	1.142E-04	1.336E-03	9.914E-03	5.733E-02	2.003E-01
RA226	5.867E-09	1.104E-08	1.814E-08	5.113E-08	1.457E-07	1.041E-06	1.145E-05	1.142E-04	1.336E-03	9.914E-03	5.733E-02	2.003E-01
TH230	9.717E-06	1.417E-05	1.865E-05	3.224E-05	5.538E-05	1.538E-04	5.564E-04	1.960E-03	7.358E-03	2.267E-02	7.368E-02	1.990E-01
TH234	1.453E-01	1.451E-01	1.452E-01	1.451E-01	1.451E-01	1.451E-01	1.451E-01	1.451E-01	1.451E-01	1.451E-01	1.451E-01	1.451E-01
PA233	1.348E-01	1.436E-01	1.436E-01	1.440E-01	1.450E-01	1.526E-01	1.914E-01	2.883E-01	4.613E-01	5.413E-01	5.435E-01	5.400E-01
PA234M	1.472E-01	1.451E-01	1.452E-01	1.451E-01	1.451E-01	1.451E-01	1.451E-01	1.451E-01	1.451E-01	1.451E-01	1.451E-01	1.451E-01
U234	5.125E-01	5.154E-01	5.183E-01	5.272E-01	5.415E-01	5.935E-01	7.231E-01	8.643E-01	9.008E-01	8.968E-01	8.821E-01	8.419E-01
U236	1.159E-01	1.159E-01	1.159E-01	1.160E-01	1.160E-01	1.161E-01	1.166E-01	1.180E-01	1.225E-01	1.337E-01	1.590E-01	1.799E-01
U238	1.451E-01	1.451E-01	1.451E-01	1.451E-01	1.451E-01	1.451E-01	1.451E-01	1.451E-01	1.451E-01	1.451E-01	1.451E-01	1.451E-01
NP237	1.403E-01	1.436E-01	1.436E-01	1.440E-01	1.450E-01	1.526E-01	1.914E-01	2.883E-01	4.613E-01	5.413E-01	5.435E-01	5.400E-01
NP239	9.236E 06	7.631E 00	7.630E 00	7.628E 00	7.625E 00	7.611E 00	7.563E 00	7.427E 00	6.971E 00	5.815E 00	3.084E 00	5.036E-01
PU238	9.832E 02	1.054E 03	1.061E 03	1.040E 03	1.001E 03	8.564E 02	4.970E 02	1.052E 02	4.867E-01	4.561E-06	6.144E-20	0.0
PU239	1.400E 02	1.424E 02	1.424E 02	1.424E 02	1.424E 02	1.423E 02	1.421E 02	1.413E 02	1.387E 02	1.314E 02	1.084E 02	6.201E 01
PU240	2.358E 02	2.358E 02	2.359E 02	2.360E 02	2.361E 02	2.363E 02	2.352E 02	2.305E 02	2.145E 02	1.747E 02	8.525E 01	1.097E 01
PU241	5.736E 04	5.470E 04	5.216E 04	4.524E 04	3.568E 04	1.380E 04	4.975E 02	4.758E-02	9.583E-03	8.103E-03	4.505E-03	8.419E-04
PU242	8.294E-01	8.295E-01	8.295E-01	8.295E-01	8.295E-01	8.294E-01	8.294E-01	8.291E-01	8.281E-01	8.251E-01	8.147E-01	7.855E-01
AM241	5.015E 01	1.397E 02	2.249E 02	4.567E 02	7.740E 02	1.474E 03	1.731E 03	1.269E 03	4.139E 02	1.685E 01	4.734E-03	8.419E-04
AM243	7.621E 00	7.631E 00	7.630E 00	7.628E 00	7.625E 00	7.611E 00	7.563E 00	7.427E 00	6.971E 00	5.815E 00	3.084E 00	5.036E-01
CM242	1.743E 04	3.713E 03	7.867E 02	9.020E 00	1.551E 00	1.412E 00	1.026E 00	4.123E-01	1.694E-02	1.859E-06	2.546E-20	0.0
CM244	7.091E 02	6.827E 02	6.570E 02	5.857E 02	4.836E 02	2.249E 02	1.541E 01	7.268E-03	1.657E-14	0.0	0.0	0.0
SUMTOT	9.313E 06	6.068E 04	5.529E 04	4.772E 04	3.833E 04	1.676E 04	3.137E 03	1.764E 03	7.847E 02	3.381E 02	2.038E 02	7.931E 01
TOTAL	1.932E 07	6.069E 04	5.530E 04	4.773E 04	3.834E 04	1.676E 04	3.140E 03	1.765E 03	7.848E 02	3.381E 02	2.040E 02	7.978E 01

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Table 2.25. Watts of light-element isotopes in a PWR assembly

	DISCHARGE	1. YR	2. YR	5. YR	10. YR	30. YR	100. YR	300. YR	1000. YR	3000. YR	10000. YR	30000. YR
C 14	6.341E-04	6.340E-04	6.339E-04	6.337E-04	6.333E-04	6.318E-04	6.265E-04	6.115E-04	5.618E-04	4.411E-04	1.891E-04	1.683E-05
CL 36	2.188E-05	2.188E-05	2.188E-05	2.188E-05	2.188E-05	2.188E-05	2.187E-05	2.186E-05	2.183E-05	2.173E-05	2.139E-05	2.046E-05
MN 54	3.704E-01	1.606E-01	6.966E-02	5.680E-03	8.710E-05	4.814E-12	0.0	0.0	0.0	0.0	0.0	0.0
FE 55	3.537E 00	2.709E 00	2.075E 00	9.326E-01	2.459E-01	1.189E-03	9.342E-12	0.0	0.0	0.0	0.0	0.0
CO 58	4.321E 01	1.240E 00	3.560E-02	8.418E-07	1.640E-14	0.0	0.0	0.0	0.0	0.0	0.0	0.0
CO 60	6.064E 01	5.315E 01	4.659E 01	3.138E 01	1.624E 01	1.164E 00	1.148E-04	4.109E-16	0.0	0.0	0.0	0.0
NI 59	1.348E-02	1.348E-02	1.348E-02	1.348E-02	1.348E-02	1.348E-02	1.347E-02	1.345E-02	1.336E-02	1.314E-02	1.236E-02	1.040E-02
NI 63	1.216E-01	1.207E-01	1.197E-01	1.171E-01	1.127E-01	9.697E-02	5.723E-02	1.268E-02	6.502E-05	1.857E-11	0.0	0.0
ZN 65	5.080E-01	1.807E-01	6.431E-02	2.897E-03	1.655E-05	1.750E-14	0.0	0.0	0.0	0.0	0.0	0.0
ZR 95	1.082E 02	2.267E 00	4.751E-02	4.372E-07	1.767E-15	0.0	0.0	0.0	0.0	0.0	0.0	0.0
NB 93M	1.169E-06	2.100E-06	2.977E-06	5.326E-06	8.439E-06	1.480E-05	1.768E-05	1.772E-05	1.772E-05	1.769E-05	1.760E-05	1.734E-05
NB 94	6.479E-03	6.479E-03	6.478E-03	6.478E-03	6.477E-03	6.472E-03	6.456E-03	6.412E-03	6.258E-03	5.839E-03	4.580E-03	2.289E-03
NB 95	1.050E 02	4.553E 00	9.997E-02	8.951E-07	3.617E-15	0.0	0.0	0.0	0.0	0.0	0.0	0.0
MO 93	3.476E-05	3.475E-05	3.474E-05	3.472E-05	3.468E-05	3.452E-05	3.397E-05	3.243E-05	2.759E-05	1.738E-05	3.449E-06	3.392E-08
IN113M	9.616E-01	1.064E-01	1.177E-02	1.594E-05	2.573E-10	1.943E-29	0.0	0.0	0.0	0.0	0.0	0.0
SN113	1.541E 00	1.705E-01	1.886E-02	2.487E-05	4.122E-10	3.113E-29	0.0	0.0	0.0	0.0	0.0	0.0
SN119M	1.626E 00	5.785E-01	2.059E-01	9.276E-03	5.304E-05	5.622E-14	0.0	0.0	0.0	0.0	0.0	0.0
SN121M	5.644E-04	5.566E-04	5.489E-04	5.265E-04	4.913E-04	3.723E-04	1.410E-04	8.798E-06	5.301E-10	4.783E-22	0.0	0.0
SN123	7.036E-01	9.892E-02	1.391E-02	3.855E-05	2.119E-09	1.932E-26	0.0	0.0	0.0	0.0	0.0	0.0
SB125	2.516E 00	1.977E 00	1.534E 00	7.160E-01	2.012E-01	1.255E-03	2.397E-11	0.0	0.0	0.0	0.0	0.0
TE125M	1.426E-01	1.287E-01	1.000E-01	4.671E-02	1.312E-02	8.185E-05	1.564E-12	0.0	0.0	0.0	0.0	0.0
SUMTOT	3.291E 02	6.747E 01	5.101E 01	3.323E 01	1.683E 01	1.284E 00	7.811E-02	3.323E-02	2.032E-02	1.947E-02	1.717E-02	1.274E-02
TOTAL	1.378E 03	6.753E 01	5.102E 01	3.323E 01	1.683E 01	1.285E 00	7.816E-02	3.325E-02	2.033E-02	1.949E-02	1.719E-02	1.275E-02

Table 2.26. Watts of fission-product isotopes in a PWR assembly

	DISCHARGE	1. YR	2. YR	5. YR	10. YR	30. YR	100. YR	300. YR	1000. YR	3000. YR	10000. YR	30000. YR
SE 79	4.700E-05	4.700E-05	4.700E-05	4.700E-05	4.699E-05	4.698E-05	4.695E-05	4.685E-05	4.650E-05	4.552E-05	4.224E-05	3.812E-05
KR 85	6.549E 00	6.141E 00	5.757E 00	4.742E 00	3.433E 00	9.432E-01	1.025E-02	2.509E-08	5.757E-28	0.0	0.0	0.0
SR 89	1.256E 03	9.655E 00	7.415E-02	3.366E-08	9.023E-19	0.0	0.0	0.0	0.0	0.0	0.0	0.0
SR 90	4.099E 01	3.999E 01	3.902E 01	3.624E 01	3.203E 01	1.956E 01	3.479E 00	2.509E-02	7.958E-10	2.992E-31	0.0	0.0
Y 90	2.013E 02	1.881E 02	1.835E 02	1.704E 02	1.507E 02	9.199E 01	1.636E 01	1.180E-01	3.743E-09	1.407E-30	0.0	0.0
Y 91	1.727E 03	2.311E 01	3.073E-01	7.220E-07	2.999E-16	0.0	0.0	0.0	0.0	0.0	0.0	0.0
ZR 93	1.591E-04	1.592E-04	1.592E-04	1.592E-04	1.592E-04	1.592E-04	1.592E-04	1.591E-04	1.590E-04	1.588E-04	1.580E-04	1.557E-04
NB 93M	1.676E-05	2.856E-05	3.970E-05	6.949E-05	1.090E-04	1.897E-04	2.261E-04	2.268E-04	2.266E-04	2.263E-04	2.252E-04	2.219E-04
ZR 95	3.481E 03	7.296E 01	1.529E 00	1.407E-05	5.685E-14	0.0	0.0	0.0	0.0	0.0	0.0	0.0
NB 95	3.363E 03	1.465E 02	3.218E 00	2.881E-05	1.164E-13	0.0	0.0	0.0	0.0	0.0	0.0	0.0
TC 99	4.118E-03	4.139E-03	4.139E-03	4.139E-03	4.138E-03	4.138E-03	4.137E-03	4.135E-03	4.125E-03	4.098E-03	4.005E-03	3.752E-03
RU106	1.539E 01	7.751E 00	3.903E 00	4.982E-01	1.613E-02	1.769E-08	2.448E-29	0.0	0.0	0.0	0.0	0.0
RH106	2.796E 03	1.275E 03	6.420E 02	8.196E 01	2.653E 00	2.911E-06	4.027E-27	0.0	0.0	0.0	0.0	0.0
AG110M	3.331E 01	1.219E 01	4.465E 00	2.191E-01	1.445E-03	2.707E-12	0.0	0.0	0.0	0.0	0.0	0.0
SB125	2.173E 01	1.701E 01	1.319E 01	6.159E 00	1.731E 00	1.078E-02	2.059E-10	0.0	0.0	0.0	0.0	0.0
SN126	2.545E-04	2.545E-04	2.545E-04	2.544E-04	2.544E-04	2.544E-04	2.543E-04	2.539E-04	2.527E-04	2.492E-04	2.374E-04	2.067E-04
SB126	9.095E 00	8.989E-04	8.989E-04	8.989E-04	8.988E-04	8.987E-04	8.983E-04	8.970E-04	8.927E-04	8.804E-04	8.387E-04	7.300E-04
SB126M	2.705E 00	4.239E-03	4.239E-03	4.239E-03	4.238E-03	4.238E-03	4.236E-03	4.230E-03	4.209E-03	4.152E-03	3.955E-03	3.443E-03
I129	8.598E-06	8.728E-06	8.728E-06	8.728E-06	8.728E-06	8.728E-06	8.728E-06	8.728E-06	8.728E-06	8.727E-06	8.724E-06	8.717E-06
CS134	7.355E 02	5.254E 02	3.753E 02	1.368E 02	2.543E 01	3.100E-02	1.830E-12	0.0	0.0	0.0	0.0	0.0
CS135	6.997E-05	7.011E-05	7.011E-05	7.011E-05	7.011E-05	7.011E-05	7.011E-05	7.010E-05	7.009E-05	7.005E-05	6.990E-05	6.948E-05
CS137	4.936E 01	4.824E 01	4.714E 01	4.399E 01	3.921E 01	2.474E 01	4.935E 00	4.933E-02	4.927E-09	4.908E-29	0.0	0.0
BA137M	1.780E 02	1.736E 02	1.697E 02	1.583E 02	1.411E 02	8.903E 01	1.776E 01	1.776E-01	1.773E-08	1.766E-28	0.0	0.0
CE144	3.703E 02	1.520E 02	6.240E 01	4.318E 00	5.037E-02	9.318E-10	0.0	0.0	0.0	0.0	0.0	0.0
PR144	4.309E 03	1.756E 03	7.210E 02	4.989E 01	5.819E-01	1.077E-08	0.0	0.0	0.0	0.0	0.0	0.0
PM147	2.233E 01	1.806E 01	1.387E 01	6.277E 00	1.675E 00	8.515E-03	7.889E-11	0.0	0.0	0.0	0.0	0.0
SM151	2.053E-02	2.073E-02	2.058E-02	2.012E-02	1.939E-02	1.670E-02	9.909E-03	2.230E-03	1.205E-05	4.010E-12	0.0	0.0
EU154	4.350E 01	4.013E 01	3.702E 01	2.906E 01	1.982E 01	3.869E 00	1.360E-02	1.343E-09	0.0	0.0	0.0	0.0
EU155	2.697E 00	2.335E 00	2.021E 00	1.310E 00	6.361E-01	3.537E-02	1.433E-06	4.047E-19	0.0	0.0	0.0	0.0
SUMTOT	1.867E 04	4.515E 03	2.325E 03	7.302E 02	4.187E 02	2.302E 02	4.258E 01	3.822E-01	1.000E-02	9.889E-03	9.540E-03	8.621E-03
TOTAL	8.702E 05	4.527E 03	2.327E 03	7.307E 02	4.188E 02	2.303E 02	4.258E 01	3.822E-01	1.001E-02	9.894E-03	9.544E-03	8.624E-03

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Table 2.27. Watts of heavy-element isotopes in a PWR assembly

	DISCHARGE	1. YR	2. YR	5. YR	10. YR	30. YR	100. YR	300. YR	1000. YR	3000. YR	10000. YR	30000. YR
BI214	8.177E-11	1.537E-10	2.526E-10	7.120E-10	2.029E-09	1.449E-08	1.595E-07	1.590E-06	1.861E-05	1.380E-04	7.982E-04	2.788E-03
PO210	3.886E-12	1.058E-11	2.160E-11	1.021E-10	5.636E-10	8.990E-09	2.042E-07	2.914E-06	4.206E-05	3.120E-04	1.804E-03	6.303E-03
PO214	3.791E-10	5.029E-10	8.265E-10	2.330E-09	6.640E-09	4.743E-08	5.219E-07	5.202E-06	6.088E-05	4.517E-04	2.612E-03	9.125E-03
PO218	2.127E-10	3.998E-10	6.570E-10	1.852E-09	5.278E-09	3.771E-08	4.149E-07	4.136E-06	4.840E-05	3.591E-04	2.077E-03	7.254E-03
RN222	1.914E-10	3.596E-10	5.910E-10	1.666E-09	4.748E-09	3.392E-08	3.732E-07	3.720E-06	4.354E-05	3.230E-04	1.868E-03	6.525E-03
RA226	1.661E-10	3.124E-10	5.134E-10	1.447E-09	4.124E-09	2.946E-08	3.242E-07	3.232E-06	3.782E-05	2.806E-04	1.623E-03	5.668E-03
TH230	2.746E-07	4.004E-07	5.269E-07	9.109E-07	1.565E-06	4.346E-06	1.572E-05	5.539E-05	2.079E-04	6.407E-04	2.082E-03	5.623E-03
U234	1.475E-02	1.483E-02	1.492E-02	1.517E-02	1.559E-02	1.708E-02	2.081E-02	2.488E-02	2.593E-02	2.581E-02	2.539E-02	2.423E-02
U236	3.143E-03	3.143E-03	3.143E-03	3.143E-03	3.144E-03	3.148E-03	3.161E-03	3.198E-03	3.320E-03	3.625E-03	4.310E-03	4.877E-03
U238	3.672E-03	3.672E-03	3.672E-03	3.672E-03	3.672E-03	3.672E-03	3.672E-03	3.672E-03	3.672E-03	3.672E-03	3.672E-03	3.672E-03
NP237	4.121E-03	4.218E-03	4.219E-03	4.229E-03	4.259E-03	4.484E-03	5.623E-03	8.469E-03	1.355E-02	1.590E-02	1.597E-02	1.586E-02
PU238	3.256E 01	3.492E 01	3.513E 01	3.445E 01	3.314E 01	2.836E 01	1.646E 01	3.483E 00	1.612E-02	1.510E-07	2.035E-21	0.0
PU239	4.351E 00	4.427E 00	4.427E 00	4.426E 00	4.426E 00	4.423E 00	4.415E 00	4.392E 00	4.310E 00	4.082E 00	3.370E 00	1.927E 00
PU240	7.344E 00	7.346E 00	7.347E 00	7.351E 00	7.356E 00	7.362E 00	7.327E 00	7.179E 00	6.682E 00	5.443E 00	2.655E 00	3.416E-01
PU241	2.380E 00	2.270E 00	2.164E 00	1.877E 00	1.480E 00	5.728E-01	2.064E-02	1.974E-06	3.976E-07	3.362E-07	1.869E-07	3.493E-08
PU242	2.448E-02	2.449E-02	2.449E-02	2.449E-02	2.449E-02	2.448E-02	2.448E-02	2.448E-02	2.445E-02	2.436E-02	2.405E-02	2.319E-02
AM241	1.674E 00	4.661E 00	7.505E 00	1.524E 01	2.583E 01	4.918E 01	5.776E 01	4.236E 01	1.381E 01	5.624E-01	1.580E-04	2.810E-05
AM243	2.782E-01	2.785E-01	2.785E-01	2.784E-01	2.783E-01	2.778E-01	2.761E-01	2.711E-01	2.544E-01	2.123E-01	1.126E-01	1.838E-02
CM242	6.422E 02	1.368E 02	2.899E 01	3.324E-01	5.715E-02	5.204E-02	3.782E-02	1.519E-02	6.242E-04	6.849E-08	9.384E-22	0.0
CM243	2.149E-01	2.103E-01	2.058E-01	1.928E-01	1.730E-01	1.122E-01	2.463E-02	3.237E-04	8.416E-11	1.290E-29	0.0	0.0
CM244	2.481E 01	2.388E 01	2.299E 01	2.049E 01	1.692E 01	7.866E 00	5.390E-01	2.583E-04	5.797E-16	0.0	0.0	0.0
SUBTOT	7.158E 02	2.149E 02	1.091E 02	8.469E 01	8.971E 01	9.826E 01	8.692E 01	5.776E 01	2.515E 01	1.038E 01	6.225E 00	2.402E 00
TOTAL	3.652E 04	2.149E 02	1.091E 02	8.471E 01	8.973E 01	9.828E 01	8.693E 01	5.777E 01	2.516E 01	1.039E 01	6.234E 00	2.417E 00

Table 2.31. Total curies in PWR assembly as a function of age

Years after discharge	Activation products	Heavy elements	Fission products	Total
0	2.070×10^{5}	1.932×10^{7}	7.293×10^{7}	9.246×10^{7}
1	9.098×10^{3}	6.069×10^{4}	1.059×10^{6}	1.129×10^{6}
2	5.762×10^{3}	5.530×10^{4}	5.673×10^{5}	6.284×10^{5}
5	3.244×10^{3}	4.773×10^{4}	2.155×10^{5}	2.665×10^{5}
10	1.586×10^{3}	3.834×10^{4}	1.425×10^{5}	1.824×10^{5}
30	3.414×10^{2}	1.676×10^{4}	8.135×10^{4}	9.845×10^{4}
100	1.482×10^{2}	3.140×10^{3}	1.534×10^{4}	1.863×10^{4}
300	3.555×10^{1}	1.765×10^{3}	1.646×10^{2}	1.965×10^{3}
1,000	3.696	7.848×10^{2}	1.001×10^{1}	7.985×10^{2}
3,000	3.321	3.381×10^{2}	9.847	3.513×10^{2}
10,000	2.800	2.040×10^{2}	9.647	2.164×10^{2}
30,000	2.079	7.978×10^{1}	9.108	9.097×10^{1}

Table 2.32. Total watts of thermal power in PWR assembly as a function of age

Years after discharge	Activation products	Heavy elements	Fission products	Total
0	1.378×10^{3}	3.652×10^{4}	8.702×10^{5}	9.081×10^{5}
1	6.753×10^{1}	2.149×10^{2}	4.527×10^{3}	4.809×10^{3}
2	5.102×10^{1}	1.091×10^{2}	2.327×10^{3}	2.487×10^{3}
5	3.323×10^{1}	8.471×10^{1}	7.307×10^{2}	8.486×10^{2}
10	1.683×10^{1}	8.973×10^{1}	4.188×10^{2}	5.254×10^{2}
30	1.285	9.828×10^{1}	2.303×10^{2}	3.299×10^{2}
100	7.816×10^{-2}	8.693×10^{1}	4.258×10^{-1}	1.296×10^{2}
300	3.325×10^{-2}	5.777×10^{1}	3.822×10^{-1}	5.819×10^{1}
1,000	2.033×10^{-2}	2.516×10^{1}	1.001×10^{-2}	2.519×10^{1}
3,000	1.949×10^{-2}	1.039×10^{1}	9.894×10^{-3}	1.042×10^{1}
10,000	1.719×10^{-2}	6.234	9.544×10^{-3}	6.261
30,000	1.275×10^{-2}	2.417	8.624×10^{-3}	2.438

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Table 2.33. Spontaneous fission neutron source in a PWR assembly (neutrons/sec)

	DISCHARGE	1. YR	2. YR	5. YR	10. YR	30. YR	100. YR	300. YR	1000. YR	3000. YR	10000. YR	30000. YR
PU238	1.459E 05	1.565E 05	1.578E 05	1.584E 05	1.485E 05	1.271E 05	7.376E 04	1.561E 04	7.224E 01	6.769E-04	9.118E-18	0.0
PU240	9.248E 05	9.250E 05	9.252E 05	9.256E 05	9.262E 05	9.270E 05	9.226E 05	9.040E 05	8.418E 05	6.854E 05	3.348E 05	4.301E 04
PU242	3.522E 05	3.522E 05	3.522E 05	3.522E 05	3.522E 05	3.522E 05	3.522E 05	3.521E 05	3.516E 05	3.504E 05	3.459E 05	3.335E 05
CM242	1.032E 08	2.198E 07	4.657E 06	5.339E 04	9.181E 03	8.360E 03	6.075E 03	2.440E 03	1.003E 02	1.100E-02	1.507E-16	0.0
CM244	9.699E 07	9.338E 07	8.987E 07	8.011E 07	6.615E 07	3.076E 07	2.107E 06	9.941E 02	2.267E-09	0.0	0.0	0.0
CM246	3.780E 05	3.779E 05	3.779E 05	3.777E 05	3.774E 05	3.763E 05	3.725E 05	3.617E 05	3.263E 05	2.431E 05	8.680E 04	4.576E 03
TOTALS												
TABLE	2.020E 08	1.172E 08	9.634E 07	8.198E 07	6.797E 07	3.255E 07	3.835E 06	1.637E 06	1.520E 06	1.279E 06	7.674E 05	3.814E 05
ACTUAL	2.020E 08	1.172E 08	9.634E 07	8.198E 07	6.797E 07	3.255E 07	3.835E 06	1.637E 06	1.520E 06	1.279E 06	7.674E 05	3.814E 05

Table 2.34. Alpha, n neutron source in a PWR assembly (neutrons/sec)

	DISCHARGE	1. YR	2. YR	5. YR	10. YR	30. YR	100. YR	300. YR	1000. YR	3000. YR	10000. YR	30000. YR
PO210	2.024E-07	5.512E-07	1.125E-06	5.318E-06	2.935E-05	4.682E-04	1.064E-02	1.518E-01	2.190E 00	1.625E 01	9.397E 01	3.282E 02
PO213	6.730E-04	1.843E-04	1.900E-04	2.106E-04	2.550E-04	5.581E-04	3.362E-03	3.097E-02	4.820E-01	5.793E 00	6.350E 01	3.676E 02
PO214	5.262E-05	6.980E-05	1.147E-04	3.234E-04	9.216E-04	6.584E-03	7.244E-02	7.221E-01	8.451E 00	6.270E 01	3.626E 02	1.267E 03
PO218	1.608E-05	3.021E-05	4.965E-05	1.400E-04	3.989E-04	2.850E-03	3.135E-02	3.125E-01	3.658E 00	2.714E 01	1.569E 02	5.482E 02
AT217	3.705E-04	1.015E-04	1.046E-04	1.160E-04	1.404E-04	3.073E-04	1.851E-03	1.705E-02	2.654E-01	3.189E 00	3.496E 01	2.024E 02
RN222	1.092E-05	2.053E-05	3.373E-05	9.510E-05	2.710E-04	1.936E-03	2.130E-02	2.123E-01	2.485E 00	1.844E 01	1.066E 02	3.724E 02
FR221	2.397E-04	6.565E-05	6.769E-05	7.502E-05	9.082E-05	1.988E-04	1.197E-03	1.103E-02	1.717E-01	2.063E 00	2.262E 01	1.309E 02
RA226	6.527E-06	1.228E-05	2.018E-05	5.688E-05	1.621E-04	1.158E-03	1.274E-02	1.270E-01	1.486E 00	1.103E 01	6.377E 01	2.228E 02
AC225	1.788E-04	4.898E-05	5.050E-05	5.597E-05	6.776E-05	1.483E-04	8.933E-04	8.231E-03	1.281E-01	1.539E 00	1.687E 01	9.769E 01
TH230	1.078E-02	1.567E-02	2.062E-02	3.564E-02	6.123E-02	1.701E-01	6.152E-01	2.167E 00	8.136E 00	2.507E 01	8.146E 01	2.200E 02
U234	6.062E 02	6.096E 02	6.131E 02	6.236E 02	6.405E 02	7.020E 02	8.554E 02	1.022E 03	1.066E 03	1.061E 03	1.043E 03	9.959E 02
U236	1.101E 02	1.101E 02	1.102E 02	1.102E 02	1.102E 02	1.103E 02	1.108E 02	1.121E 02	1.164E 02	1.270E 02	1.511E 02	1.709E 02
U238	1.072E 02	1.072E 02	1.072E 02	1.072E 02	1.072E 02	1.072E 02	1.072E 02	1.072E 02	1.072E 02	1.072E 02	1.072E 02	1.072E 02
NP237	1.787E 02	1.829E 02	1.830E 02	1.834E 02	1.847E 02	1.945E 02	2.439E 02	3.673E 02	5.878E 02	6.897E 02	6.925E 02	6.880E 02
PU238	1.439E 06	1.543E 06	1.552E 06	1.522E 06	1.464E 06	1.253E 06	7.272E 05	1.539E 05	7.122E 02	6.673E-03	8.990E-17	0.0
PU239	1.538E 05	1.565E 05	1.565E 05	1.564E 05	1.564E 05	1.563E 05	1.560E 05	1.552E 05	1.523E 05	1.483E 05	1.191E 05	6.811E 04
PU240	2.522E 05	2.523E 05	2.523E 05	2.525E 05	2.526E 05	2.528E 05	2.516E 05	2.466E 05	2.295E 05	1.869E 05	9.120E 04	1.173E 04
PU241	1.421E 05	1.355E 05	1.293E 05	1.121E 05	8.841E 04	3.421E 04	1.233E 03	1.179E-01	2.375E-02	2.008E-02	1.116E-02	2.086E-03
PU242	8.253E 02	8.254E 02	8.254E 02	8.254E 02	8.254E 02	8.253E 02	8.253E 02	8.250E 02	8.240E 02	8.211E 02	8.107E 02	7.816E 02
AM241	1.018E 05	2.834E 05	4.564E 05	9.268E 05	1.571E 06	2.991E 06	3.512E 06	2.576E 06	8.399E 05	3.420E 04	9.608E 00	1.709E 00
AM243	2.145E 04	2.148E 04	2.148E 04	2.147E 04	2.146E 04	2.142E 04	2.129E 04	2.091E 04	1.962E 04	1.637E 04	8.681E 03	1.818E 03
CM242	1.052E 08	2.242E 07	4.751E 06	5.447E 04	9.366E 03	8.529E 03	6.198E 03	2.490E 03	1.023E 02	1.122E-02	1.538E-16	0.0
CM243	1.688E 04	1.652E 04	1.616E 04	1.515E 04	1.359E 04	8.814E 03	1.935E 03	2.583E 01	6.611E-06	1.014E-24	0.0	0.0
CM244	3.768E 06	3.628E 06	3.491E 06	3.112E 06	2.570E 06	1.195E 06	8.186E 04	3.862E 01	8.805E-11	0.0	0.0	0.0
TOTALS												
TABLE	1.111E 08	2.846E 07	1.083E 07	6.175E 06	6.149E 06	5.923E 06	4.762E 06	3.157E 06	1.245E 06	3.848E 05	2.229E 05	8.800E 04
ACTUAL	1.111E 08	2.846E 07	1.083E 07	6.175E 06	6.149E 06	5.923E 06	4.762E 06	3.157E 06	1.245E 06	3.848E 05	2.229E 05	8.800E 04

PART 2

NUCLEAR WASTES FROM DECOMMISSIONING AND DISMANTLING OF NUCLEAR REACTORS

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ABSTRACT

This report is a brief review of the nuclear wastes that arise from the decommissioning and dismantling (D&D) of nuclear reactors. The principal emphasis is on the quantity and composition of D&D wastes that may be considered high activity low level waste. These wastes constitute a small part of the overall reactor D&D wastes although they are of importance in regard to waste management. The primary waste components of interest here are the reactor vessel and the reactor internals since these are the largest source of radioactivity, may also contain long lived activation products, and cannot be easily decontaminated.

CONTENTS

ABSTRACT iii

TABLES vi

FIGURES. vii

ACKNOWLDGEMENT viii

EXECUTIVE SUMMARY. ix

1. INTRODUCTION 1

2. U. S. DECOMMISSIONING EXPERIENCE 1

3. DECOMMISSIONING AND DISMANTLING WASTE GENERATED AT THE ELK RIVER REACTOR 9

4. DECOMMISSIONING AND DISMANTLING WASTES FROM LARGE COMMERCIAL POWER REACTORS 10

5. REFERENCES 21

TABLES

2.1 Commercial Reactors Which Soon May Be Candidates for Decommissioning 1

2.2 Decommissioned Reactors 2

2.3 Decommissioned Demonstration Nuclear Power Plants 2

2.4 Dismantled Critical Facilities (License Terminated) 3

2.5 Dismantled Research Reactors (License Terminated) 4

2.6 Decommissioned Power, Test and Nuclear Ship Reactors in Safe Storage/Mothballed With Continued License 6

2.7 Decommissioned Research Reactors and Critical Facilities in Safe Storage with Continued Possession Only License 7

2.8 Number of Nuclear Facilities Decommissioned in Different States . . 8

3.1 D&D Radioactive Waste From Elk River 9

3.2 Distribution of Radionuclides in Elk River and Their Equilibrium Fraction 10

4.1 Chemical Composition of 304-Stainless Steel, Wt. Percent 11

4.2 Composition of ASTM 533 Grade B Alloy Steel Plate, Wt. Percent . . 11

4.3 Comparison of AIF and Elk River Scale-up Waste Projections 12

4.4 Radioactivity Levels in Major Activated Reactor Components at Time of Reactor Shutdown (PWR) 16

4.5 Summary of Estimated Radioactivity Levels in Activated Steel Components (PWR) 17

4.6 Reference Radionuclide Inventory 1, Neutron-Activated Stainless Steel (BWR) at Time of Reactor Shutdown 18

4.7 Reference Radionuclide Inventory 2, Neutron-Activated Carbon Steel (BWR) at Time of Reactor Shutdown 19

4.8 Estimated Total Radioactivity in Neutron-Activated Components (BWR) 20

FIGURES

4.1 Simplified PWR Cross Section 13

4.2 Flux at Core Mid-Height (PWR), (E <0.625 eV) 15

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EXECUTIVE SUMMARY

This inventory of high level nuclear waste in the U.S. was developed from existing data, reports, and other sources of information. Existing and projected inventories of high level waste (HLW), transuranic (TRU) waste, spent fuel, and decommissioning and dismantling (D&D) waste will be required in making prudent decisions in the licensing of high level waste repositories.

Numerous references, reports, and data sources were consulted for this work. Discussions with responsible individuals at the major waste storage facilities were held to help resolve and understand discrepancies that appear in the literature. Low level wastes are excluded from this inventory. For decommissioning and dismantling wastes, no detailed projection of future wastes could be developed although rough D&D waste projections for commercial power reactors are available.

The inventory is reported in three parts. Part 1 gives the existing and projected inventory of HLW, spent fuel, and TRU wastes. Part 2 is concerned with the characteristics and volume of high activity D&D waste from commercial power reactors. Part 3 provides additional information on D&D wastes from power reactors and particle accelerators.

The major findings of the reports may be summarized as follows:

- A reasonable estimate of the total inventory of existing HLW and commercial spent fuel is available in terms of the volume, average range of chemical composition, and radioactivity.

- The accuracy of certain inventory estimates and the methods used to estimate the quantitites and characteristics of the waste are not all known at this time. There are also certain discrepancies that cannot be resolved without additional study and measurement.

- There are about 75 million gallons and about 1300 million curies of liquid sludge, salt cake, and other forms of HLW at the government facilities. About 40% of the total volume of HLW is in a liquid form. The Idaho, Savannah River, and Hanford facilities contain, respectively, about 3%, 30%, and 67% of the total waste volume and about 7%, 43%, and 45% of the waste radioactivity in curies. The total waste and the percentages among the three varies from year to year, and the data given above are for 1978.

- A considerable number of irradiated fuel elements awaiting processing are stored at government facilities. The exact number and characteristics of all these elements were not determined for this inventory.

- There are about 5200 metric tons of heavy metal (MTHM) in the form of spent fuel assemblies stored at commercial power reactors and

away-from-reactor storage facilities at the end of 1979. Based upon 5.3 x 10^5 Ci/MT for 5-year-old fuel, the estimated activity would be in the range of 2500-3000 million curies.

- The most reasonable projection of nuclear electric generating capacity reaches about 180 GWE by the year 2000. This will result in an accumulation of about 89,000 MTHM in the form of spent fuel assemblies in the year 2000.

- There were about 319,000 m^3 of TRU waste buried at DOE sites as of October 1979. The quantity of TRU elements contained in this waste is estimated to be about 785 kg. About 55,000 m^3 of TRU waste is retrievably stored at DOE facilities. This waste contains about 685 kg of TRU elements.

- High activity D&D wastes from commercial nuclear power reactors are primarily neutron activated reactor components such as the reactor vessel and internals. For a large 1175 MWe PWR, the total mass of such components is estimated to be about 600 tonnes and the radioactivity level about 5 x 10^6 curies. For a BWR of 1155 MWe, the mass of activated components estimated to be about 300 tonnes and the radioactivity level about 6.5 x 10^6 curies. These wastes are important in regard to waste management since they may contain long lived activation products and cannot be easily decontaminated.

- Only rough estimates are available for D&D wastes from other nuclear facilities, such as particle accelerators. Decommissioning and dismantling waste volumes from surplus DOE facilities are reported to be approximately 9.5 x 10^3 m^3 of TRU wastes and 1.1 x 10^5 m^3 of non-TRU wastes by the year 2000. However, most of these are believed to be low activity wastes.

The characteristics of HLW and TRU wastes, such as the chemical composition, physical form, and radionuclide content, are important in analyzing the behavior of candidate waste forms for long term isolation. When appropriate, the characterization of HLW and TRU wastes should be determined with greater certainty than is now available.

1. INTRODUCTION

This report is a brief examination of the nuclear wastes that arise from the decommissioning and dismantling (D&D) of nuclear reactors. An overall review of D&D experience in the United States is given in the next section. This is followed by a brief discussion of the waste generated by the D&D of the Elk River reactor. The last section describes the nuclear wastes that may be generated by the D&D of large commercial power reactors.

The principal emphasis in this report is on the quantity and composition of D&D wastes that may be considered high activity low level waste. These wastes constitute a small part of the overall waste that is generated in the D&D of a nuclear reactor although they are of importance in regard to waste management. NRC is currently developing guidelines for classifying wastes as to their ultimate disposal, although this effort is not yet completed. The DOE currently classifies D&D waste as either TRU or non-TRU waste. In this report we are therefore primarily concerned with the quantity and composition of activated reactor components such as the reactor vessel and reactor internals. These components are of interest because they are the largest source of radioactivity, may also contain long lived activation products, and cannot be easily decontaminated. While these wastes may be broadly classified as low level wastes, because of their activities and comparatively long half lives, special treatment and disposal methods may be warranted.

2. U. S. DECOMMISSIONING EXPERIENCE

The U.S. commercial nuclear program is about twenty years old. The Shippingport Nuclear Power Station started in 1957. The Dresden Unit 1 was placed in operation in 1960 and the Yankee in 1961. These are the reactors that would be the object of the first D&D activities. It will probably be near the end of this century before any actual decommissioning is done. Table 2.1 gives a list of six commercial reactors which may be considered for decommissioning[1] at the end of this century. The total capacity of these reactors is only 865 MWe.

Table 2.1

Commercial Reactors Which Soon May Be Candidates for Decommissioning[1]

Reactor	Type	Size (MWe)	Year of Start Up
1. Shippingport Nuclear Power Station	PWR	90	1957
2. Dresden Nuclear Power Station, Unit 1	BWR	200	1959
3. Yankee Nuclear Power Station	PWR	175	1960
4. Big Rock Point Nuclear Plant	BWR	72	1962
5. Indian Point Station, Unit 1	PWR	265	1962
6. Humbolt Bay Power Plant, Unit 3	BWR	63	1963

The actual D&D experience in the U.S. is rather limited. So far, 68 reactors or nuclear facilities have been decommissioned.[2] These include four demonstration nuclear power plants, five power reactors, six test reactors and 34 research reactors. The breakdown is shown in Table 2.2. The thermal power, location and present status of the decommissioned demonstration nuclear power plants are given in Table 2.3.[2] Similar data for 17 dismantled critical facilities are shown in Table 2.4.[2] Note that these critical facilities did not have much power, usually being in the range of 100 watts. Table 2.5 gives the list of the 25 dismantled research reactors.[2] Tables 2.6 and 2.7 show, respectively, the decommissioned power and research reactors in safe storage. The nuclear ship Savannah has been placed in safe storage with no definite date for dismantlement.

Table 2.2

Decommissioned Reactors

	Number
Demonstration Nuclear Power Plants	4
Power Reactors	5
Test Reactors	6
Research Reactors	34
Critical Facilities	18
Nuclear Ship Reactor	1
TOTAL	68

Table 2.3

Decommissioned Demonstration Nuclear Power Plants[2]

Docket No. Reactor	Thermal Power	Location	Present Status
115-1 Elk River BWR	58.2 MW	Elk River Minnesota	Dismantled, Federal control terminated.
115-2 Piqua Organic Cooled	45.5 MW	Piqua Ohio	Entombed, DOE monitoring.
115-3 Hallam Sodium Cooled	256 MW	Hallam Nebraska	Entombed, DOE monitoring.
115-4 Bonus BWR Nuclear Superheat	50 MW	Rincon Puerto Rico	Entombed, DOE monitoring.

Table 2.4

Dismantled Critical Facilities (License Terminated)[2]

Docket No. Reactor	Max. Power	Location	Date Lic. Terminated
50-13 Babcock & Wilcox (Split Table)	1 KW	Lynchburg Virginia	06-01-73
50-14 Battelle Memorial Platics Moderated Critical Assembly	200 W	W. Jefferson Ohio	05-11-70
50-23 Nuclear Development Corp. of America (Crit. Ex.)	100 W	Pawling New York	06-22-61
50-24 General Electric (BWR Crit. Ex.)	200 W	Alameda Co. California	12-01-69
50-34 Westinghouse Electric Corp. (PWR Crit. Ex.)	1 KW	Waltz Mill Pennsylvania	12-08-69
50-37 Gen. Dynamics (CIRGA Zirconium Hydride Mod.)	25 W	San Diego California	03-15-60
50-75 NASA (ZPR-1, Solution Type Crit. Fac.)	100 W	Cleveland Ohio	10-13-73
50-87 Westinghouse Electric Corp. (Crit. Ex. Station)	100 W	Waltz Mill Pennsylvania	01-26-72
50-108 Allis Chalmers (Crit. Ex. Fac.)	100 W	Greendale Wisconsin	01-20-67
50-153 Westinghouse Electric Corp. (CVTR MOCKUP, Heavy Water)	3 KW	Waltz Mill Pennsylvania	04-24-63
50-154 Martin Marietta Corp. (Liquid Fluidized Bed Crit. Ex.)	10 W	Middle River Maryland	02-07-66
50-191 Babcock & Wilcox (Plutonium Recycle Crit. Ex.)	-	Lynchburg Virginia	06-01-73
50-197 NASA (ZPR02 Solution Type Crit. Fac.)	100 W	Cleveland Ohio	10-13-73
50-203 GE (Mixed Spectrum Crit. Assembly)	400 W	Alameda Co. California	03-11-68
50-234 Gulf Oil Corp. (Thermionic Crit. Fac.)	200 W	San Diego California	08-10-73
50-246 General Dynamics Corp. ACRE	10 KW	San Diego California	12-30-66
50-290 Gulf United Nuclear (Water Mod. Proof Test Fac.)	100 W	Pawling New York	06-25-74

Table 2.5

Dismantled Research Reactors (License Terminated)[2]

Docket No. Reactor	Thermal Power	Location	Date Lic. Terminated
50-1 Illinois Inst. of Technology (Water Boiler Research)	100 KW	Chicago Illinois	04-28-72
50-4 USN Research Lab (Pool Type)	1 MW	Washington D.C.	03-18-71
50-8 N.C. State (Aqueous Homogeneous)	100 W	Raleigh N. Carolina	09-07-66
50-17 Industrial Reactor Labs. (Pool Type)	5 MW	Plainsboro New Jersey	11-04-77
50-43 U.S. Naval Post-graduate School (AGN-201)	0.1 W	Monterey California	10-11-72
50-50 North American Aviation (L-47 homogeneous)	5 W	Canoga Park California	06-30-58
50-58 Oklahoma State University (AGN-201)	0.1 W	Stillwater Oklahoma	03-19-74
50-60 U.S. Navy Hospital (AGN-201M)	5 W	Bethesda Maryland	06-24-65
50-64 University of Akron (AGN-201)	0.1 W	Akron Ohio	10-09-67
50-84 University of Calif. (AGN-201)	0.1 W	Berkeley California	08-23-66
50-98 University of Delaware (AGN-201)	0.1 W	Newark Delaware	02-26-79
50-101 Gulf United Nuclear (Pawling Lattice Test Rig)	100 W	Pawling New York	06-25-74
50-114 William March Rice University (AGN-211)	15 W	Houston Texas	09-26-67
50-122 University of Wyoming (L-77)	10 W	Laramie Wyoming	12-05-75

Table 2.5 (Continued)

Dismantled Research Reactors (License Terminated)[2]

Docket No. Reactor	Thermal Power	Location	Date Lic. Terminated
50-135 Walter Reed Medical Center (L-54, Homogeneous Solution)	50 KW	Washington D.C.	07-26-72
50-167 Lockheed (Pool Type)	10 W	Dawson Co. Georgia	09-01-60
50-172 Lockheed (Radiation Effects Reactor)	3 MW	Dawson Co. Georgia	08-31-71
50-202 University of Nevada (L-77)	10 W	Reno Nevada	02-24-75
50-212 General Dynamics Fast Critical Assembly	500 W	San Diego California	03-05-65
50-216 Polytechnic Inst. N.Y. (AGN-201M)	0.1 W	Bronx New York	12-21-77
50-227 General Atomic Co. (TRIGA Mark III)	1.5 MW	San Diego California	12-10-75
50-235 Gulf General Atomic (APFA)	500 W	San Diego California	10-22-69
50-240 Gulf General Atomic (HTGR)	100 W	San Diego California	04-02-73
50-253 Gulf Oil Corp. (APFA III)	500 W	San Diego California	08-10-73
50-310 NUMEC and Commonwealth of Pa. (Pool)	1 MW	Quehanna Pennsylvania	12-02-66

Table 2.6

Decommissioned Power, Test and Nuclear Ship Reactors in Safe Storage/Mothballed With Continued License[2]

Docket No. Reactor	Thermal Power	Location	Present Status
50-16 Fermi 1 Fast Breeder Power Reactor	200 MW	Monroe Co. Michigan	Possession only lic.
50-18 GE VBWR BWR Power Reactor	50 MW	Alameda Co. California	Possession only lic.
50-114 CVTR Pressure Tube, Heavywater, Power Reactor	65 MW	Parr S. Carolina	Byproduct lic. (St.)
50-130 Pathfinder Nuclear Superheat, Power Reactor	190 MW	Sioux Falls S. Dakota	Byproduct lic. (NRC)
50-171 Peach Bottom 1 HTGR Power Reactor	115 MW	York Co. Pennsylvania	Possession only lic.
50-22 Westinghouse Test Reactor	60 MW	Waltz Mill Pennsylvania	Possession only lic.
50-30 NASA Plum Brook Test Reactor	60 MW	Sandusky Ohio	Dismantling plans being developed
50-146 Saxton PWR Test Reactor	28 MW	Saxton Pennsylvania	Possession only lic.
50-183 GE EVESR Exp. Superheat Test Reactor	17 MW	Alameda Co. California	Possession only lic.
50-200 B&W BAWTR Test Reactor (Pool Type)	6 MW	Lynchburg Virginia	Byproduct lic. (NRC)
50-231 SEFOR Sodium Cooled Test Reactor	20 MW	Strickler Arkansas	Byproduct lic. (St.)
50-238 NS Savannah PWR	80 MW	Charleston S. Carolina	Possession only lic.

Table 2.7

Decommissioned Research Reactors and Critical Facilities in Safe Storage With Continued Possession Only License[2]

Docket No. Reactor	Thermal Power	Location	Present Status
50-6 Battelle Memorial Institute Pool Type Research Reactor	2 MW	Columbus Ohio	Dismantling plans being developed.
50-47 Watertown Arsenal U.S. Army Pool Type Research Reactor	5 MW	Watertown Massachusetts	Dismantling plans being developed.
50-94 Rockwell Inter. Corp. L-77 Research Reactor	10 W	Canoga Park California	Dismantling authorized.
50-106 Oregon State Univ. AGN-201 Research Reactor	0.1 W	Corvallis Oregon	Dismantling authorized.
50-111 North Carolina State Pool Type Research Reactor	10 KW	Raleigh N. Carolina	Dismantling plans being developed.
50-129 West Virginia Univ. AGN-211P Research Reactor	75 W	Morgantown W. Virginia	Dismantling plans being developed.
50-141 Stanford Univ. Pool Type Research Reactor	10 KW	Stanford California	Dismantling authorized.
50-147 Rockwell Int. Corp. FCEL Split Table Critical Facility	200 W	Canoga Park California	Dismantling authorized.
50-185 NASA MOCKUP Pool Type Research Reactor	100 KW	Sandusky Ohio	Dismantling plans being developed.
50-384 Calif. Polytechnic State Univ. AGN-201 Research Reactor	0.1 W	San Luis Obispo California	Dismantling plans being developed.

The number of nuclear reactors or facilities decommissioned in different states to date is shown in Table 2.8. It will be seen there that the largest number of nuclear facilities decommissioned is in California, totaling 19. This is followed by Ohio, Pennsylvania, and New York, where only eight, seven, and four facilities, respectively, have been so far decommissioned.

Table 2.8

Number of Nuclear Facilities Decommissioned in Different States

State	Demonstration Nuclear Power Plants	Nuclear Reactors and Critical Facilities in Safe Storage With Continued/Possession Only License	Dismantled Research Reactors	Dismantled Critical Facilities	Total Number
Arkansas		1			1
California		6	8	5	19
Delaware			1		1
Georgia			2		2
Illinois			1		1
Maryland			1	1	2
Massachusetts		1			1
Michigan		1			1
Minnesota	1				1
Nebraska	1				1
Nevada			1		1
New Jersey			1		1
New York			2	2	4
North Carolina		1	1		2
Ohio	1	3	1	3	8
Oklahoma			1		1
Oregon		1			1
Pennsylvania		3	1	3	7
Puerto Rico	1				1
South Carolina		2			2
South Dakota		1			1
Texas			1		1
Virginia		1		2	3
Washington, D.C.			2		2
West Virginia		1			1
Wisconsin				1	1
Wyoming			1		1
TOTAL	4	22	25	17	68

Most of the reactors of interest to us, except for the Elk River Reactor, were either mothballed or entombed so that relavant information on D&D waste generation cannot be directly obtained from these reactors. The best record of D&D wastes comes from the Elk River Reactor dismantlement work. This is described in the next section.

3. DECOMMISSIONING AND DISMANTLING WASTE GENERATED AT THE ELK RIVER REACTOR

The Elk River Reactor in Minnesota was a small reactor with 58.2 MWt capacity. It was operated only for 3.5 years. The D&D wastes generated from this reactor cannot, therefore, be easily extrapolated to the commercial size reactors of today. Particularly, the designs for the flux levels of the power reactors will differ considerably from the Elk River Reactor. However, it is important to understand the radioactive wastes generated by the Elk River Plant so that future D&D wastes by power reactors can be predicted with some degree of confidence. The mass, volume and gross activity of the radioactive wastes produced by the D&D activity of the Elk River Plant are shown in Table 3.1.[3,4] These are given according to different locations of the waste materials, namely, the reactor pressure vessel, internal materials, external materials, biological shield and miscellaneous. As expected, the internal materials contribute most to the radioactivity in the D&D wastes, about 8×10^3 Ci/m^3 or about 1 Ci/Kg. The activities of the miscellaneous wastes are not given, probably because these were low.

Table 3.1

D&D Radioactive Waste From Elk River[3,4]

Component	Mass (Kg)	Volume (m^3)	Activity (Ci)
Reactor Pressure Vessel	36,000	4.6	1,100
Reactor Internals	8,100	1.1	8,700
Externals	54,000	5.3	440
Biological Shield	39,000	5.9	5.8
Miscellaneous Radioactive & Contaminated Materials	1.1×10^6	1,300	Not Available
Miscellaneous Contaminated Concrete	3.0×10^6	1,300	Not Available

A distribution of the calculated activities of the important nuclides in the Elk River waste is shown in Table 3.2. These calculations took into account the actual operating history of the Elk River. The total activity for these eight nuclides is only 9800 curies. This should be compared to over six million curies of radioactivity expected from decommissioning of a large 1150 MWE BWR (see Table 4.8). The eight nuclei shown in Table 3.2 constitute 99% of the total activity in Elk River D&D waste.

Table 3.2

Distribution of Radionuclides in Elk River and Their Equilibrium Fraction

Nuclide	Half-Life (yrs)	Activity Ci
V-49	0.90	2.3 E0
Mn-54	0.86	3.3 E1
Fe-55	2.70	5.6 E3
Co-57	0.74	2.0 E0
Co-60	5.30	3.9 E3
Ni-63	100	2.9 E2
Sb-125	2.7	2.2 E0
Eu-152	13	2.1 E0
TOTAL		9.8 E3

4. DECOMMISSIONING AND DISMANTLING WASTES FROM LARGE COMMERCIAL POWER REACTORS

The radionuclide inventory present at shutdown in a defueled power reactor is due to activated components, activated corrosion products and fission products that adhere to piping or are deposited in filters. From a waste management point of view, the activated reactor vessel and internals are of greater concern. This is because these components represent the largest source of radioactivity, may also contain long lived activation products, and cannot be easily decontaminated. Moreover, the NRC is currently developing guidelines for classifying radioactive wastes according to requirements for their safe disposal as described in NUREG-0456.[4] While this classification system is not yet complete, it appears likely that certain activated reactor components may not be suitable for disposal in shallow land burial sites.

The radioactivity generated in the reactor vessel and other internal components comes about primarily by neutron capture in the materials used for the reactor vessel and internal components. Various internal components such as the core barrel, thermal shield, and upper and lower core plates in PWRs are manufactured of Type 304 stainless steel. The materials contained in 304-SS are shown in Table 4.1.[5]

Table 4.1

Chemical Composition of 304-Stainless Steel,
Wt. Percent

Carbon	.08 (max)
Manganese	2.00 (max)
Phosphorous	.045 (max)
Sulphur	.03 (max)
Silicon	1.00 (max)
Chromium	18.00-20.00
Nickel	8.00-12.00
Iron	65-70

Reactor pressure vessels are generally fabricated from various alloy steels. For example, the Dresden-1, Humboldt Bay, and Big Rock Point reactor vessels are made of A302 Grade B alloy steel. The Donald C. Cook PWR pressure vessel is fabricated from ASTM A-533 Grade B steel. The pressure vessels also have a stainless steel cladding on the inside of the vessel. The composition of ASTM 533 Grade B is shown in Table 4.2.[6]

Table 4.2

Composition of ASTM 533 Grade B
Alloy Steel Plate, Wt. Percent

Carbon	.25
Manganese	1.15-1.50
Phosphorous	.035
Sulphur	.04
Silicon	.15-.30
Nickel	.40-.70
Molybdenum	.45-.60
Iron	95.0 -97.0

Various other steel alloys are used in fabricating reactor internals but the materials described above are typical.

For each element present in a given material, there may be several isotopes of that element present. For example, for nickel, the isotopic abundances for the isotopes of interest are Ni-58 about 68% and for Ni-62 about 3.6%. Each isotope has a different neutron capture cross section and yields a different activation product. The common activation products in 304-SS are Fe-55, Ni-63, Ni-59, and Cr-51. Other impurities, such as small amounts of cobalt which occur naturally as an impurity in iron and nickel, give rise to other activation products such as Co-60 .

The most common mechanism for the production of activation products is the n,γ reaction in which the parent isotope captures a neutron. Other reactions, such as n,2n and n,p reactions are less common and contribute considerably less to the production of activation products. The production of C-14 is an exception to this; most is produced from an n,p reaction in N-14.

The masses of the various reactor internal components and the reactor pressure vessel are dependent on the size and design of the reactor. Estimates of the volumes and masses of these various components can be obtained by using data supplied by the reactor vendor or the safety analysis report for the plant under study. A scale-up technique using Elk River data has been reported in reference 4. Table 4.3, taken from reference 4, compares the results of an Elk River scale-up approach by Ford Bacon Davis Utah to the results of an Atomic Industrial Forum Study of 1976.[7] There appears to be general agreement among these results and the estimates provided below.

Table 4.3

Comparison of AIF and Elk River Scale-Up Waste Projections

	Mass (Kg)			Activity (at 5 years)(Ci)		
	AIF-PWR 3411 MW	AIF-BWR 3579 MW	FB & DU 3600 MW	AIF-PWR 3411 MW	AIF-BWR 3579 MW	FB & DU 3600 MW
Reactor Pressure Vessel	3.8×10^5	3.0×10^5	9.8×10^5	2.1×10^4	4.7×10^3	5.1×10^4
Internals	1.6×10^5	9.7×10^4	2.2×10^5	5.0×10^6	5.1×10^5	4.0×10^5
Externals			1.5×10^6			2.0×10^4
Biological Shield	3.7×10^4	6.2×10^5	1.0×10^6	3.5×10^2	4.7	2.7×10^2

To estimate the radioactivity from activated reactor internals it is necessary to have an estimate of the neutron flux that the component has been subjected to. Flux levels vary considerably throughout the core and pressure vessel, however, maximum flux levels may be used to estimate the upper bound on the radioactivity of the activated components of interest. The mid-height position of the core is the location where the neutron flux is at a maximum axially so that activity estimates using these values are conservative. For a large PWR the following neutron flux levels are typical:[8]

Inside surface of core barrel - 2×10^{12} n/cm^2-sec
Inside surface of thermal shield - 6×10^{11} n/cm^2-sec
Cladding and inside surface of reactor pressure vessel - 8×10^{10} n/cm^2-sec.

Figure 4.1 is a simplified schematic of the core and reactor vessel cross section for a typical PWR. An approximate plot of the neutron flux (core

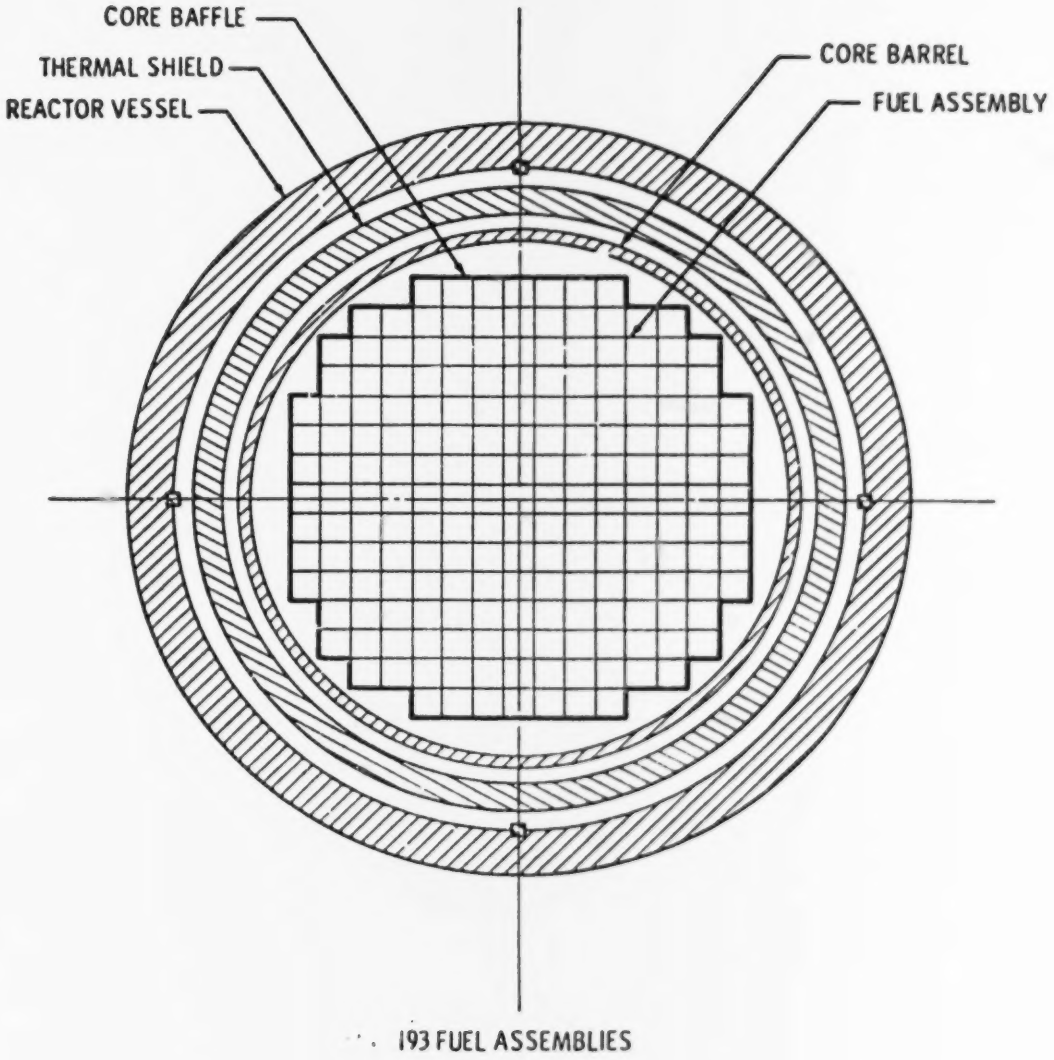

Figure 4.1. Simplified PWR cross section.[8]

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mid-height position) for a typical PWR is shown in Figure 4.2. These Figures use data contained in the Final Safety Analysis Report for the Donald C. Cook Nuclear Plant.[8]

Calculation of the buildup of activation products in the reactor internals is generally done using computer codes such as the ORIGEN code. In addition, computer codes for characterizing the neutron flux levels at various locations are also utilized. However, a rough estimate of the activity at shutdown can be obtained from the following equation:

$$\text{Activity } (Ci/cm^3) = N\sigma\phi(1-e^{-\lambda T})/3.7\times10^{10}$$

where

N = atomic density of the parent isotope (atoms/cm^3)

σ = the thermal neutron cross section of the parent isotope (cm^2)

ϕ = neutron flux (n/cm^2-sec)

λ = the decay constant of the activation product (yr^{-1})

T = irradiation time (yr).

The radioactivity levels in activated reactor components were computed for a reference PWR in reference 9 using computer codes for the flux distributions and buildup of activation products. The reference reactor is a large state-of-the-art Westinghouse 1175 MWE PWR such as the TROJAN plant. The results of these calculations are shown in Table 4.4 for each major activation product. A summary of these data is given in Table 4.5. These results are based on 30 effective full power years (EFPY) of reactor operation. The compositions of the vessel and internal components are based on nominal handbook values as described above. The isotopic abundances for each element are those that occur in nature.

Similar results for a reference 1155 MWE General Electric BWR plant for 30 EFPY of operation are reported in reference 10. The reference reactor is based on data for the Washington Public Power Supply System (WPPSS) Nuclear Project No. 2 (WNP-2) reactor. Table 4.6, which is taken from reference 10, gives the reference radionuclide inventory for irradiated stainless steel. Table 4.7 gives similar results for carbon steel which is the material used for reactor pressure vessels. The data in these tables represent the maximum values present at shutdown. Table 4.8 gives the estimated radioactivity in each of the major neutron-activated components in the reference BWR. The data on the mass of each component was calculated from the estimated volume by using a density of 8×10^3 kg/m^3 for the steel components.

The exposure dose rate resulting from the radioactivity in activated reactor internals has been estimated and is reported in references 9 and 10. While the total radioactivity in certain components does not decrease rapidly in about 100 years following shutdown, the dose rate does decrease significantly. This is due in part to the fact that Ni-63 controls the total radioactivity after the first ten years while Co-60 controls the exposure dose rate.

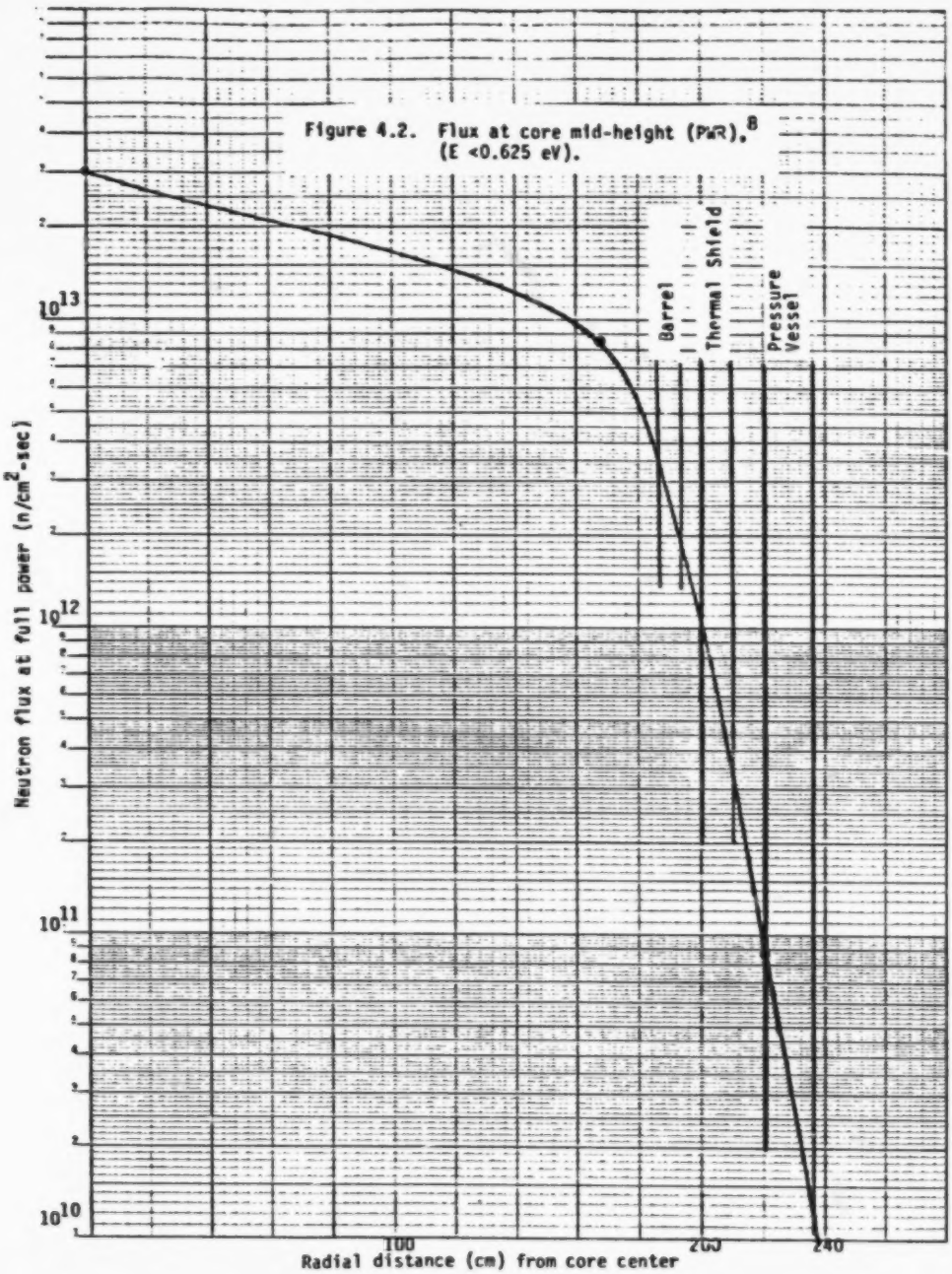

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Table 4.4

Radioactivity Levels in Major Activated Reactor Components at Time of Reactor Shutdown (PWR) [9]

Isotope	Half-Life	Core Mid-Plane Radioactivities (Ci/m^3) Shroud	Lower 4.72 m of Core Barrel	Thermal Shields	Vessel Inner Cladding	Lower 5.02 m of Vessel Wall	Upper Grid Plate[a]	Lower Grid Plate[a]
^{95}Nb	35 day	2.0×10^{3}	7.6×10^{0}	3.5×10^{0}	5.6×10^{-3}	1.7×10^{-3}		
^{59}Fe	45 day	4.6×10^{4}	4.4×10^{3}	2.0×10^{3}	1.0×10^{2}	2.7×10^{1}		
^{58}Co	72 day	1.5×10^{5}	1.0×10^{4}	4.6×10^{3}	3.3×10^{2}	6.6×10^{0}		
^{95}Zr	65 day	1.1×10^{-1}	6.2×10^{-3}	2.9×10^{-3}	2.0×10^{-4}	7.2×10^{-4}		
^{65}Zn	245 day	1.2×10^{2}	1.1×10^{0}	5.0×10^{-1}	6.7×10^{-4}	3.5×10^{-5}		
^{54}Mn	~300 day	6.8×10^{4}	3.7×10^{3}	1.7×10^{3}	1.2×10^{2}	4.7×10^{1}		
^{55}Fe	2.7 yr	1.3×10^{6}	1.5×10^{5}	6.7×10^{4}	3.5×10^{3}	7.2×10^{2}		
^{60}Co[b] upper	5.27 yr	9.6×10^{5}	9.3×10^{4}	4.7×10^{4}	2.5×10^{3}	7.5×10^{1}		
lower		3.2×10^{5}	3.1×10^{4}	1.6×10^{4}	8.2×10^{2}	2.5×10^{1}		
^{63}Ni	~100 yr	1.2×10^{5}	1.5×10^{4}	6.8×10^{3}	3.6×10^{2}	3.8×10^{0}		
^{93}Mo	~3000 yr	3.6×10^{-1}	5.2×10^{-2}	2.4×10^{-2}	1.2×10^{-3}	1.3×10^{-3}		
^{14}C	~5,750 yr	1.5×10^{2}	1.8×10^{1}	8.3×10^{0}	4.0×10^{-1}	1.9×10^{-2}		
^{94}Nb	~20,000 yr	5.4×10^{0}	2.6×10^{-1}	1.2×10^{-1}	9.5×10^{-3}	--		
^{59}Ni	~80,000 yr	7.4×10^{2}	1.3×10^{2}	5.0×10^{1}	3.0×10^{0}	3.2×10^{-2}		
Sum (Ci/m^3)		2.97×10^{6}	3.07×10^{5}	1.45×10^{5}	7.73×10^{3}	9.04×10^{2}	2.97×10^{6}	2.97×10^{6}
Average/Peak		0.755	0.637	0.778	0.637	0.637	0.003×4.74[c]	0.08×4.74[c]
Ci/kg[d]		2.787×10^{2}	2.433×10^{1}	1.403×10^{1}	7.164×10^{-1}	7.164×10^{-2}	5.254×10^{0}	1.403×10^{2}
Weight of Material (kg)		12,312	26,783	10,413	2,074	245,582	4,627	3,946
Sum (Ci)[e]		3.431×10^{6}	6.516×10^{5}	1.461×10^{5}	1.581×10^{3}	1.759×10^{4}	2.431×10^{4}	5.534×10^{5}

Total - Radioactivity

4.826×10^{6} Curies
1.786×10^{17} becquerels

(a) Normalized to shroud

(b) Upper and lower bounds were computed using the maximum and minimum levels of ^{59}Co contaminant in the materials. All totals were computed using the upper bound values.

(c) $\frac{\text{Activity (Plate Average)}}{\text{Activity (Plate Edge)}} = 4.74$, $\frac{\text{Activity (Shroud at Place Location)}}{\text{Activity (Shroud at Axial Midplane)}} = \begin{cases} 0.005, \text{ upper plate} \\ 0.08, \text{ lower plate} \end{cases}$

(d) Conversion factor assumes stainless steel density of 8.038×10^{3} kg/m^3 (0.29 $lb/in.^3$).

(e) The number of significant figures carried is for computational accuracy and does not imply precision to four places.

Table 4.5

Summary of Estimated Radioactivity Levels in Activated Steel Components (PWR)

Component	Estimated Radioactivity (Ci)	Mass, tonnes
Pressure Vessel, Cylindrical Wall	19,171	269.8
Vessel Head	<10	88.45
Vessel Bottom	<10	38.9
Upper Core Upper Support Columns	<100	11.11
Upper Core Barrel	<1000	
Upper Core Grid Plate	24,310	4.63
Guide Tubes	<100	15.10
Lower Core Barrel	651,600	45.36[a]
Thermal Shields	146,100	10.43
Core Shroud	3,431,000	12.3
Lower Core Grid Plate	553,400	3.95
Lower Support Columns	<10,000	3.36
Lower Core Forging	<2,500	36.47
Misc. Internals	<2,000	36.29
Reactor Cavity Liner	<10	
TOTAL	<4,841,321	588.13

[a] Includes upper core barrel mass.

Table 4.6

Reference Radionuclide Inventory I,
Neutron-Activated Stainless Steel[a] (BWR)[10]
at Time of Reactor Shutdown

Radionuclide	Radioactivity Concentration at Shutdown (Ci/m^3)	Fractional Radioactivity at Shutdown
^{3}H	4.65×10^{-4}	1.63×10^{-10}
^{10}Be	2.63×10^{-6}	---[b]
^{14}C	1.05×10^{2}	3.68×10^{-5}
^{32}P	1.11×10^{2}	3.89×10^{-5}
^{33}P	6.65×10^{2}	2.33×10^{-4}
^{35}S	5.52×10^{1}	1.94×10^{-5}
^{36}Cl	2.69×10^{-4}	---[b]
^{51}Cr	1.45×10^{6}	5.09×10^{-1}
^{54}Mn	8.50×10^{3}	2.98×10^{-3}
^{55}Fe	9.22×10^{5}	3.24×10^{-1}
^{59}Fe	2.74×10^{4}	9.61×10^{-3}
^{58}Co	2.10×10^{4}	7.37×10^{-3}
^{60}Co	3.36×10^{5}	1.18×10^{-1}
^{59}Ni	6.36×10^{2}	2.23×10^{-4}
^{63}Ni	8.75×10^{4}	3.07×10^{-2}
^{65}Zn	3.23×10^{1}	1.13×10^{-5}
^{93}Zr	8.15×10^{-6}	---[b]
^{95}Zr	1.41×10^{-2}	4.21×10^{-5}
^{93m}Nb	1.35×10^{-1}	4.74×10^{-8}
^{94}Nb	1.50×10^{0}	5.26×10^{-7}
^{95}Nb	1.20×10^{2}	4.21×10^{-5}
^{93}Mo	3.26×10^{-1}	1.14×10^{-7}
^{99}Tc	3.18×10^{-2}	1.12×10^{-8}
^{108m}Ag	7.36×10^{-2}	2.58×10^{-8}
^{108}Ag	8.67×10^{1}	3.04×10^{-5}
^{109m}Ag	3.61×10^{0}	1.27×10^{-6}
^{109}Cd	3.42×10^{0}	1.20×10^{-6}
^{110m}Ag	8.04×10^{0}	2.82×10^{-6}
^{110}Ag	2.02×10^{2}	7.09×10^{-5}
^{151}Sm	2.12×10^{-2}	7.44×10^{-9}
^{152}Eu	1.12×10^{-3}	3.93×10^{-10}
^{154}Eu	3.12×10^{0}	1.09×10^{-6}
^{160}Tb	9.48×10^{-4}	3.33×10^{-10}
^{166m}Ho	7.84×10^{-4}	2.75×10^{-10}
TOTAL	2.85×10^{6}	1.00

[a] Calculated at the inner surface of the 304 stainless steel core shroud, at the axial midplane of the fuel zone, for 30 EFPY of operation.

[b] Indicates a value of less than 1.00×10^{-10}.

Table 4.7

Reference Radionuclide Inventory II,
Neutron-Activated Carbon Steel[a] (BWR)[10]
at Time of Reactor Shutdown

Radionuclide	Radioactivity Concentration at Shutdown (Ci/m^3)	Fractional Radioactivity at Shutdown
^{14}C	6.77×10^{-3}	1.84×10^{-5}
^{32}P	9.20×10^{-1}	2.51×10^{-3}
^{35}S	1.76×10^{-2}	4.80×10^{-5}
^{51}Cr	1.75×10^{0}	4.77×10^{-3}
^{54}Mn	1.06×10^{1}	2.89×10^{-2}
^{55}Fe	3.35×10^{2}	9.13×10^{-1}
^{59}Fe	9.44×10^{0}	2.57×10^{-2}
^{58}Co	1.49×10^{0}	4.06×10^{-3}
^{60}Co	6.49×10^{0}	1.77×10^{-2}
^{59}Ni	1.46×10^{-2}	3.98×10^{-5}
^{63}Ni	1.73×10^{0}	4.71×10^{-3}
^{65}Zn	1.88×10^{-6}	5.12×10^{-9}
^{98}Sr	5.24×10^{-13}	1.43×10^{-15}
^{91}Y	2.25×10^{-13}	6.13×10^{-16}
^{95}Zr	1.62×10^{-4}	4.41×10^{-7}
^{93m}Nb	3.90×10^{-4}	1.06×10^{-6}
^{94}Nb	8.30×10^{-7}	2.26×10^{-9}
^{95}Nb	3.76×10^{-4}	1.02×10^{-6}
^{93}Mo	9.39×10^{-4}	2.56×10^{-6}
^{99}Tc	2.64×10^{-4}	7.19×10^{-7}
TOTAL	3.67×10^{2}	1.00

[a] Calculated at the inner surface of the SA 533 carbon steel reactor vessel, at the axial midplane of the fuel zone, for 30 EFPY of operation.

Table 4.8

Estimated Total Radioactivity in Neutron-Activated Components (BWR)[10]

Component (quantity)	Estimated Activated Volume (m^3)	Radioactivity per Component (Ci)	Estimated Total Radioactivity (Ci)	Estimated mass, tonnes
Core Shroud (1)	3.75	6.30×10^6	6.30×10^6	30.0
Jet Pump Assembly (10)	0.076	2.00×10^3	2.00×10^4	.6
Reactor Vessel (1)				
Cladding	0.428	4.58×10^2	2.16×10^3	3.4
Shell Wall*	15.26	1.70×10^3		122.0
Sacrificial Shield (1)				
Inner Shell	2.19	1.03×10^2		17.5
Reinforced Concrete Region	73.30	3.47×10^0	1.66×10^2	--
Outer Shell	6.22	5.39×10^1		49.7
Steam Separator Assembly (1)				
Shroud Head Plate	0.841	8.65×10^3	9.60×10^3	6.7
Steam Separator Risers	0.376	9.52×10^2		3.0
Top Fuel Guide (1)	0.310	3.01×10^4	3.01×10^4	
Orificed Fuel Support (193)	0.0036	3.63×10^0	7.01×10^2	22.8
Core Support Plate (1)	2.54	6.50×10^2	6.50×10^2	
Incore Instrument Strings (55)	0.00026	1.99×10^2	1.10×10^4	
Control Rod (185)	0.0019	9.61×10^2	1.78×10^5	.04
Control Rod Guide Tube (185)	0.0024	5.12×10^{-1}	9.47×10^1	
TOTAL			6.55×10^6	258.0

*Total mass of vessel including top head, studs, nuts, and washers is 748 tonnes.

Ni-63 has a half life of about 100 years, but its primary radiation (67 keV beta) is not energetic enough to penetrate the outer layers of the skin. This is of course important in regard to personnel exposure during decommissioning activities. The dose to man from other pathways, such as contamination of drinking water or uptake in the food chain, primarily from disposal of such wastes has not been thoroughly examined. The purpose of this brief report is to provide a perspective on the quantities and characteristics of activated reactor components which will probably represent the largest inventory of such wastes from the commercial nuclear fuel cycle.

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PART 3

NUCLEAR WASTES FROM DECOMMISSIONING AND DISMANTLING OF NUCLEAR FACILITIES

H. K. Manaktala

ABSTRACT

Waste quantity projections of the high activity radioactive wastes anticipated from the decommissioning and dismantling (D&D) of principal nuclear facilities, viz. commercial power plants, particle accelerators, and surplus DOE facilities, are provided. The major emphasis has been on identifying the principal sources of radwastes, and on quantifying the wastes in terms of volume, mass, radioactive isotopes, and activity levels (curie contents).

CONTENTS

ABSTRACT iii
FIGURES vi
TABLES vii
ACKNOWLEDGEMENTS viii
EXECUTIVE SUMMARY. ix

1. INTRODUCTION 1

2. TYPES OF NUCLEAR FACILITIES. 1

3. DECOMMISSIONING ALTERNATIVES FOR A NUCLEAR FACILITY. 2

4. D&D OF NUCLEAR POWER REACTORS. 2
   4.1 Sources of D&D Radwaste 3
   4.2 Radionuclides of Concern. 3
   4.3 Radioactive Waste Inventory 7
      4.3.1 Neutron Activated Wastes 7
      4.3.2 Other Radioactive Wastes 9
   4.4 D&D Radwaste Projections. 9

5. PARTICLE ACCELERATORS. 19
   5.1 Types of Accelerators 22
   5.2 Sources of D&D Radwaste 22
   5.3 Radionuclides of Concern. 22
   5.4 Radioactive Waste Inventory 22
   5.5 Discussion. 25

6. CONCLUSIONS. 25

7. REFERENCES 27

APPENDIX A - List of Commercial Power Reactors in the U. S. (Presently Operating or Expected to be On-Line by 1994) 30

APPENDIX B1 - List of Particle Accelerators in the U. S. 36

APPENDIX B2 - List of Decommissioned Accelerators. 47

APPENDIX C - List of Propulsion Reactors (Maritime) 49

FIGURES

4.1	Sources of D&D Wastes in Nuclear Reactors	5
4.2	Decay Behavior of the Activated Inventory	6
4.3	BWR Pressure Vessel and Internals	10
4.4	PWR Pressure Vessel and Internals	11
4.5	Projection of the Cumulative Number of Commercial Power Reactors in the United States	17
4.6	Projection of the Number of Commercial Power Reactors Undergoing D&D Activities	18
4.7	Projection of the Cumulative Mass of Neutron Activated D&D Wastes From Commercial Power Reactors	20
4.8	Projection of Radioactivity Inventory From Decommissioned Commercial Power Reactors	21
5.9	Induced Activity in Selected Accelereators	26

TABLES

3.1 Decommissioned Power, Demonstration, and Test Reactors 4

4.2 Typical Weight Percent Trace Elements in Type 304 SS 7

4.3 Typical Radionuclides of Concern in Neutron Activated Materials . . 8

4.4 Reference Radionuclide Inventory 1, Neutron Activated Stainless Steel (BWR) at Time of Reactor Shutdown 12

4.5 Reference Radionuclide Inventory 2, Neutron Activated Carbon Steel at Time of Reactor Shutdown 13

4.6 Estimated Total Radioactivity in Neutron Activated Components (BWR) 14

4.7 Summary of Estimated Radioactivity Levels in Activated Steel Components (PWR) 15

4.8 Radioactivity Levels in Major Activated Reactor Components at Time of Reactor Shutdown (PWR) 16

5.9 Radioactivity Expected in Three Categories of Components From Various Types of Accelerators 23

5.10 Induced Radioactivity From Electron Beams 23

5.11 Mass of Radioactive Waste From Selected Accelerators 24

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EXECUTIVE SUMMARY

This inventory of high level nuclear waste in the U.S. was developed from existing data, reports, and other sources of information. Existing and projected inventories of high level waste (HLW), transuranic (TRU) waste, spent fuel, and decommissioning and dismantling (D&D) waste will be required in making prudent decisions in the licensing of high level waste repositories.

Numerous references, reports, and data sources were consulted for this work. Discussions with responsible individuals at the major waste storage facilities were held to help resolve and understand discrepancies that appear in the literature. Low level wastes are excluded from this inventory. For decommissioning and dismantling wastes, no detailed projection of future wastes could be developed although rough D&D waste projections for commercial power reactors are available.

The inventory is reported in three parts. Part 1 gives the existing and projected inventory of HLW, spent fuel, and TRU wastes. Part 2 is concerned with the characteristics and volume of high activity D&D waste from commercial power reactors. Part 3 provides additional information on D&D wastes from power reactors and particle accelerators.

The major findings of the reports may be summarized as follows:

- A reasonable estimate of the total inventory of existing HLW and commercial spent fuel is available in terms of the volume, average range of chemical composition, and radioactivity.

- The accuracy of certain inventory estimates and the methods used to estimate the quantitites and characteristics of the waste are not all known at this time. There are also certain discrepancies that cannot be resolved without additional study and measurement.

- There are about 75 million gallons and about 1300 million curies of liquid sludge, salt cake, and other forms of HLW at the government facilities. About 40% of the total volume of HLW is in a liquid form. The Idaho, Savannah River, and Hanford facilities contain, respectively, about 3%, 30%, and 67% of the total waste volume and about 7%, 43%, and 45% of the waste radioactivity in curies. The total waste and the percentages among the three varies from year to year, and the data given above are for 1978.

- A considerable number of irradiated fuel elements awaiting processing are stored at government facilities. The exact number and characteristics of all these elements were not determined for this inventory.

- There are about 5200 metric tons of heavy metal (MTHM) in the form of spent fuel assemblies stored at commercial power reactors and

away-from-reactor storage facilities at the end of 1979. Based upon 5.3 x 10^5 Ci/MT for 5-year-old fuel, the estimated activity would be in the range of 2500-3000 million curies.

- The most reasonable projection of nuclear electric generating capacity reaches about 180 GWE by the year 2000. This will result in an accumulation of about 89,000 MTHM in the form of spent fuel assemblies in the year 2000.

- There were about 319,000 m^3 of TRU waste buried at DOE sites as of October 1979. The quantity of TRU elements contained in this waste is estimated to be about 785 kg. About 55,000 m^3 of TRU waste is retrievably stored at DOE facilities. This waste contains about 685 kg of TRU elements.

- High activity D&D wastes from commercial nuclear power reactors are primarily neutron activated reactor components such as the reactor vessel and internals. For a large 1175 MWe PWR, the total mass of such components is estimated to be about 600 tonnes and the radioactivity level about 5 x 10^6 curies. For a BWR of 1155 MWe, the mass of activated components estimated to be about 300 tonnes and the radioactivity level about 6.5 x 10^6 curies. These wastes are important in regard to waste management since they may contain long lived activation products and cannot be easily decontaminated.

- Only rough estimates are available for D&D wastes from other nuclear facilities, such as particle accelerators. Decommissioning and dismantling waste volumes from surplus DOE facilities are reported to be approximately 9.5 x 10^3 m^3 of TRU wastes and 1.1 x 10^5 m^3 of non-TRU wastes by the year 2000. However, most of these are believed to be low activity wastes.

The characteristics of HLW and TRU wastes, such as the chemical composition, physical form, and radionuclide content, are important in analyzing the behavior of candidate waste forms for long term isolation. When appropriate, the characterization of HLW and TRU wastes should be determined with greater certainty than is now available.

1. INTRODUCTION

This report surveys the published information on the high activity radioactive wastes that would result from the decommissioning and dismantling (D&D) of power reactors, and also provides projections of volumes of transuranic (TRU) and non-transuranic (non-TRU) contaminated wastes anticipated to result from the D&D of surplus DOE nuclear facilities. The projections for waste inventories have been provided for the years through 2000 and beyond.

The major emphasis in this report has been on identifying the principal sources of high activity D&D radwaste, and on quantifying the wastes in terms of volume, radioactive isotopes, activity levels (curie contents), etc. Waste projections provided in this report are based on published data. It is neither the intent of this report to generate D&D radwaste data, nor to suggest alternatives for decommissioning a particular nuclear facility.

2. TYPES OF NUCLEAR FACILITIES

Consideration of decommissioning and dismantling (D&D) covers a broad range of nuclear facilities. Major categories of classifications are as follows:

- Power reactors (commercial)
- Nuclear ships and submarines
- Space propulsion reactors (nuclear rocket engine reactors)
- Military (weapons) facilities
- Uranium mills
- Research reactors and critical facilities
- Mixed oxide (MOX) fuel fabrication plants
- Production reactors (Pu production for military use)
- Particle accelerators

The responsibility for licensing of these facilities lies mainly with the Department of Energy (DOE), the Nuclear Regulatory Commission (NRC), or the individual states.

The DOE is responsible for monitoring and disposing of radioactivity in the facilities it owns, leases and controls. These include facilities such as research and production reactors, and accelerators located both on nonfederal and on DOE owned property - commonly referred to as DOE reservations.

The NRC is responsible for regulating private use of nuclear materials. This responsibility covers nuclear power plants, uranium mills, facilities which make or process nuclear fuel, and the regulation of users of source, byproduct, and special nuclear materials.

The licensing of particle accelerators (other than DOE owned or operated) is under the jurisdiction of the individual states in which they are located.

This report mainly discusses power reactors, production reactors, and particle accelerators, since these are the principal facilities which will generate most of the high activity D&D solid wastes.

3. DECOMMISSIONING ALTERNATIVES FOR A NUCLEAR FACILITY

Decommissioning refers to the orderly disposition of nuclear facilities after their retirement from service, taking into account the environmental, radiation protection, waste management and safety considerations. The activities involved in a decommissioning excercise can vary from simple closure of the facility with a minimum removal of radioactive materials, to complete plant disassembly and the restoration of the site for unrestricted release for other uses.

According to the NRC classification, there are three principal decommissioning alternatives or modes. These are:[1,2]

- SAFSTOR - fix and maintain property, so that the risk to safety is acceptable for the period of storage, followed by decontamination and/or decay to an unrestricted level.
- ENTOMB - encase and maintain property in a storage and structurally long lived material (e.g., concrete) to assure retention until radioactivity decays to an unrestricted level.
- DECON - immediately remove all radioactive material to permit unrestricted release of the property.

Past D&D efforts have provided experience in each of these modes. For example, a number of research and production reactors at DOE sites are currently mothballed and under surveillance (SAFSTOR mode). These mothballed facilities include the Heavy Water Components Test Reactor (HWCTR) at Savannah River, the Plutonium Recycle Test Reactor (PRTR) at Hanford, and the Experimental Boiling Water Reactor (EBWR) at Argonne National Laboratory.

Methods of entombment (ENTOMB mode) have been applied to the power demonstration reactors at Hallam, Nebraska; Piqua, Ohio, and the BONUS reactor in Puerto Rico.

The Elk River reactor in Minnesota was completely dismantled and all radioactive materials shipped offsite (DECON mode). This is the largest reactor (58.2 MWt capacity) that has been decommissioned and dismantled in the United States to date. The site is now in unrestricted use.

Combinations of the above mentioned three alternatives are also possible, such as mothballing or entombment followed by removal/dismantlement after significant radioactive decay time (50 to 100 years) (SAFSTOR/ENTOMB-DECON sequence). Such a combination of methods has the advantage of reducing the radioactive inventory and the radiation exposure risk to the workers at the time of dismantling.

4. D&D OF NUCLEAR POWER REACTORS

Although the volume of the D&D waste from a reactor will essentially remain constant with time, its radioactivity will decay. Since different decommission-

ing modes take different amounts of time for completion, the actual inventory of radwaste (in curies) that will eventually have to be properly disposed of will depend upon the decommissioning alternative chosen.

Since the majority of power reactors operating today will not be candidates for decommissioning until well after the year 2000 (assuming 40 years as the average operating life of a power reactor), decisions regarding their decommissioning timetable have not yet been made. The selection of D&D plans will be greatly influenced by the NRC regulations, governing D&D operations, in force at the time when a particular reactor is being retired from service. The economics of various D&D alternatives will also undoubtedly play a major role in the final selection of the D&D mode. All these variables make it difficult to estimate D&D wastes with any degree of confidence. Nevertheless, an attempt has been made in this report to provide a rough idea of the D&D wastes that would result from power reactors.

Table 3.1 provides a list of power, demonstration, and test reactors which have already been decommissioned. Reactor location, type, thermal rating, decommissioning mode, present status of license, monitoring system, and present storage measures are also given. A complete list of presently operating commercial power reactors in the United States, and those planned to come on-line by the year 1994 is given in Appendix A.

4.1 Sources of D&D Radwaste

The sources of radioactivity in reactor decommissioning wastes are shown in Figure 4.1

By far the greatest contribution (in terms of curie content) comes from the activation of the pressure vessel and core structural components due to neutron bombardment. Figure 4.1 also lists typical materials used for core components and the pressure vessel.

The second category of radwaste results from parts that are not directly irradiated by the neutrons, but have neutron activated corrosion products deposited in or on them, e.g. activated corrosion products in the primary heat transfer loop pipes, and contamination deposits elsewhere in the system due to fission product release from the failed fuel rods, e.g. recirculation pumps, etc.

The third category of radwaste comes from the contamination of the surfaces of facility components or structures, e.g. HEPA* filter cartridges, pool or cavity liners, ventilation/electrical duct work, surfaces of buildings and structures, soil regions, plastic sheeting and clothing, and other miscellaneous items. These wastes are classified as low level (low activity) non-TRU contaminated wastes.

4.2 Radionuclides of Concern

In the activated metallic internal parts of a light water reactor (LWR), the radionuclides Fe-55, Co-60, and Ni-63 are of principal concern after the final

*HEPA = High Efficiency Particulate Air filter.

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Table 3.1

Decommissioned Power, Demonstration and Test Reactors[3]

Reactor facility and location	Reactor type	Reactor thermal rating, MW	Type of decontaminating	Status of license	Monitoring system	Protective storage measures
CVTR, Parr, S. C.	Pressure tube, heavy water	65	Mothballing **	By-product per 10 CFR 30	Periodic surveillance	Welded closure, locked doors, security fence
Pathfinder, Sioux Falls, S. D.	BWR, nuclear superheat	190	Mothballing with steam plant conversion	By-product to state†	Continuous security force‡	Welded closure, security fence
Fermi 1, Lagonna Beach, Mich.	Sodium cooled, fast	200	Mothballing	Possession only §	Continuous security force‡	Locked doors, security fence
Peach Bottom 1, Peach Bottom, Pa.	Gas cooled, graphite moderated	115	Mothballing	Possession only	Continuous security force‡	Not yet established
VBWR, Pleasanton, Calif.	BWR	50	Mothballing with steam plant conversion	Possession only	Continuous security force‡	Locked doors, security fence
NASA Plum Brook, Sandusky, Ohio	Light water	0.1	Mothballing	Possession only	Continuous security force‡	Locked doors, security fence
GE EVESR, Pleasanton, Calif.	BWR with nuclear superheat	17	Mothballing	Possession only	Continuous security force‡	Locked doors, security fence
Saxton, Saxton, Pa.	PWR	23.5	Mothballing	Possession only	Intrusion alarms	Welded closure, locked doors, security fence
SEFOR, Sinckler, Ark.	Sodium cooled, fast	20	Mothballing	By-product to state†	Intrusion alarms	Welded closure, locked doors, security fence
Westinghouse Test Reactor, Waltz Mill, Pa	Tank	60	Mothballed	Possession only	Continuous security force‡	Locked doors, security fence
Babcock & Wilcox, Lynchburg, Va.	Pool	6	Partial dismantling	By-product per 10 CFR 30	Not required	Not required
Hallam, Hallam, Nebr.	Sodium cooled, graphite moderated	256	Entombing	Operating authorization terminated	Not required	Welded closure, concrete cover, weatherproofed
Piqua, Piqua, Ohio	Organic cooled and moderated	45.5	Entombing	Operating authorization terminated	Not required	Welded closure, concrete cover, waterproofed
BONUS, Punta Higuera, Puerto Rico	BWR with nuclear superheat	50	Entombing	Operating authorization terminated	Not required	Welded closure, concrete cover, locked doors, security fence
Elk River, Elk River, Minn.	BWR	58.2	Dismantling and partial conversion	Operating authorization terminated	Not required	Not required

*Data partially from "Decommissioning and Decontamination of Licensed Reactor Facilities and Demonstration Nuclear Power Plants" by P. B. Erickson and G. Lear, NRC, paper presented at Conference on Decontamination and Decommissioning at Idaho Falls, Idaho, August 19–21, 1975.

†A by-product license can be issued by an Agreement State per 10 CFR 150.

‡The use of a continuous security force was not required by the NRC because continuous manned security was provided for other on-site activities that were unrelated to the decommissioned reactor. If such a security force was not present, the NRC may have stipulated manned security or other additional access control measures.

§A possession only license permits possession of a reactor facility but not its operation.

**Same as SAFSTOR mode.

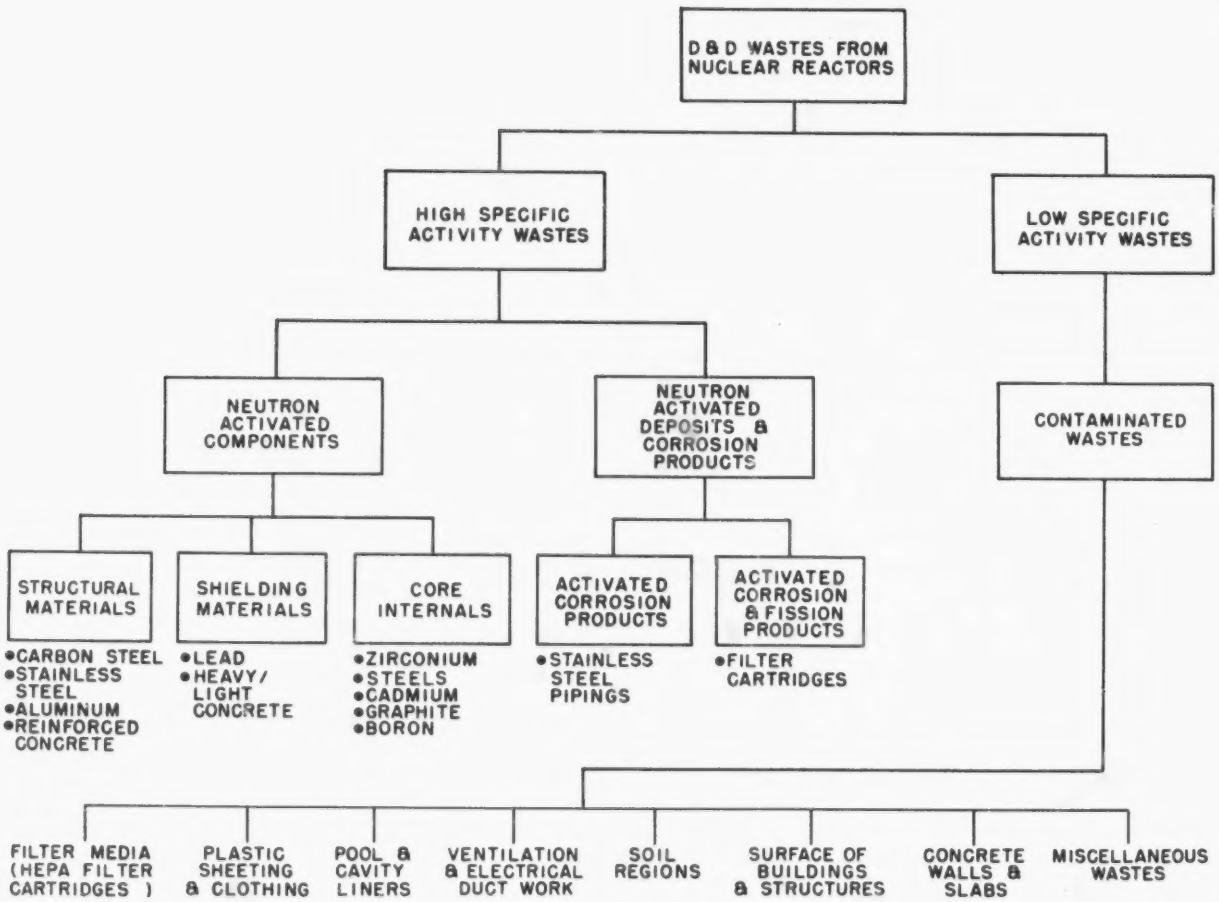

Figure 4.1. Sources of D&D wastes in nuclear reactors.

208 BEST DOCUMENT AVAILABLE

shutdown following 40 years of estimated operating life. Because of the differing radioactive half-lives of the nuclides, Fe-55 initially predominates with respect to radioactivity. After seven years of decay time, however, its radioactivity is exceeded by that of Co-60; after two more years the radioactivity of the cobalt is in turn exceeded by that of Ni-63. Figure 4.2 shows schematically the relative activities of these major radionuclides in the D&D wastes as a function of time after reactor shutdown.

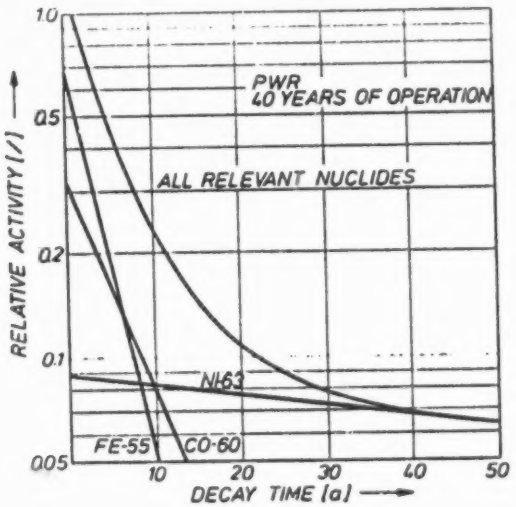

Figure 4.2. Decay behavior of the activated inventory.[4]

In planning decommissioning activities, however, Co-60 which is a significant gamma emitter is of primary concern. The dose levels of Co-60 gamma radiation from carbon steel, stainless steel cladding and internal components of the reactor vessel will determine the amount of remote operation and worker shielding that is required.

Other isotopes of concern are Ni-59, Nb-94, and C-14. Nickel-59 has not been of importance in previous decommissionings because the reactors had not operated long enough to create significant quantities of this isotope. However, a commercial Light Water Reactor (LWR) after 40 years of operation will contain significant amounts of Ni-59 in certain vessel internals. Carbon-14 is of concern primarily in High Temperature Gas-cooled Reactors (HTGR), and in graphite moderated production reactors (like those at Hanford).

Also of importance, but not well documented is the activation of trace elements that may give rise to significant activity levels and thereby affect the selection of the decommissioning mode[5]. The trace elements are present either as impurities or are intentionally added to improve certain characteristics of the structural material. For example, niobium is added to steel to improve the corrosion resistance of welds and heat affected areas. Table 4.2 lists typical weight percent trace elements in Type 304 stainless steel commonly used for reactor pressure vessels and internals. Table 4.3 is a more complete list of radionuclides of concern (as well as their emissions and emission energy levels

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that could be expected to be present in reactor pressure vessels and various other structural members as a result of neutron irradiation.

Table 4.2

Typical Weight Percent Trace Elements in Type 304 SS[6]

Element	Wt %	Remarks
Lead (Pb)	0.002	
Tin (Sn)	0.01	
Cobalt (Co)	0.20	Can be reduced on Special Order
Copper (Cu)	0.15	
Zinc (Zn)	0.01	Removed by vaporization/ oxidation
Niobium (Nb)	0.01	Also called Columbium- removed by vaporization/ oxidation
Arsenic (As)	0.01	
Aluminum (Al)	0.01	
Molybdenum (Mo)	0.05	
Boron (B)	0.0005	Removed by vaporization/ oxidation
Titanium (Ti)	0.05	Removed by vaporization/ oxidation
Vanadium (V)	0.05	Removed by vaporization/ oxidation
Selenium (Se)	0.02	
Antimony (Sb)	0.01	
Nitrogen (N)	0.04	Range is 0.03 to 0.05 in electric furnace
Oxygen (O)	0.015	
Phosphorus (P)	0.03	Can go to 0.035
Sulfur (S)	0.015-0.025	Max is 0.03
Silicon (Si)	0.60/1.0 max	Added to heat
Manganese (Mn)	1.25/2.0 max	Added to heat
Hydrogen (H)	0.0007	
Carbon (C)	0.03/0.08 max	Depending on grade

4.3 Radioactive Waste Inventory

4.3.1 Neutron Activated Wastes

The radioactive inventory in a reactor component is basically a function of material composition, trace elements present, neutron flux and spectrum to which is was exposed, irradiation time, and the decay time before the inventory was taken.

To obtain an accurate inventory, a complete operating history of the reactor, the design of the pressure vessel and its internals and their material composition must be known precisely. Since no two reactors are expected to have exactly the same operating histories, design features, and materials of construction, the process of obtaining a radioactive inventory involves making elaborate calculations for each individual reactor seperately. Computer programs and routines are available for making such calculations.[6]

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Table 4.3

Typical Radionuclides of Concern in Neutron Activated Materials[6]

	Isotope	Half-Life T 1/2 (yr)	Means of Production	Emission	Energy(MeV)
Carbon & Stainless Steel	14C	5730.0	$N^{14}(n,p)$	β^-	0.156
	49V	0.906	$Cr^{52}(p,\alpha)$	γ,β^+	0.8*
	54Mn	0.856	$Fe^{56}(d,\alpha)$	γ	0.835
	55Fe	2.8	$Fe^{54}(n,\gamma)$	γ	0.23*
	59Ni	8×10^4	$Ni^{58}(n,\gamma)$	ϵ	1.06*
	63Ni	100.0	$Ni^{62}(n,\gamma)$	β^-	0.066
	65Zn	0.667	$Zn^{64}(n,\gamma)$	γ,ϵ,β^+	1.115, 1.352, 0.325
	58Co	0.194	$Mn^{55}(\alpha,n)$	β^+,γ	0.474, 0.810
	60Co	5.263	$Co^{59}(n,\gamma)$	β^-,γ,γ	0.314, 1.17, 1.33
	93Mo	3.5×10^3	$Mo^{92}(n,\gamma)$	ϵ	Nb X-RAYS
	94Nb	2×10^4	$Nb^{93}(n,\gamma)$	β^-,γ,γ	0.49, 0.702, 0.871
	95Nb	0.096	Zr^{95}DECAY	β^-,γ	0.16, 0.765
	95Zr	0.175	$Zr^{94}(n,\gamma)$	β^-,γ,γ	0.396, 0.724, 0.756
Concrete	14C	5730.0	$N^{14}(n,p)$	β^-	0.156
	35S	0.238	$S^{34}(n,\gamma)$	β^-	0.167
	36Cl	3.01×10^5	$Cl^{35}(n,\gamma)$	β^-,ϵ	0.714, 1.18*
	37Ar	0.0953	$^{36}Ar(n,\gamma)$	ϵ	0.81*
	39Ar	269.0	$Ar^{38}(n,\gamma)$	β^-	0.565
	40K	1.28×10^9	-	β^-,γ	1.314, 1.46
	41Ca	8×10^4	$Ca^{40}(n,\gamma)$	γ	K X-RAYS
	45Ca	0.446	$Ca^{44}(n,\gamma)$	β^-	0.257
	46Sc	0.229	$Sc^{45}(n,\gamma)$	$\beta^-,\beta^-,\gamma,\gamma$	1.48, 0.357, 0.889, 1.12
	54Mn	0.856	$Fe^{56}(d,\alpha)$	γ,ϵ	0.835, 0.829, 1.379
	55Fe	2.8	$Fe^{54}(n,\gamma)$	γ	0.23*
	59Fe	0.122	$Fe^{58}(n,\gamma)$	β^-,γ,γ	1.57, 1.1, 1.29
	58Co	0.194	$Mn^{55}(\alpha,n)$	β^+,γ	0.474, 0.81
	60Co	5.263	$Co^{59}(n,\gamma)$	β^-,γ,γ	0.314, 1.17, 1.33
	59Ni	8×10^4	$Ni^{58}(n,\gamma)$	ϵ	1.06*
	63Ni	100.0	$Ni^{62}(n,\gamma)$	β	0.067
	65Zn	0.667	$Zn^{64}(n,\gamma)$	γ,ϵ,β^+	1.115, 1.352, 0.325
	94Nb	2×10^4	$Nb^{93}(n,\gamma)$	β^-,γ,γ	0.49, 0.702, 0.871
	95Nb	0.096	Zr^{95}DECAY	β^-,γ	0.16, 0.765
	93Mo	3.5×10^3	$Mo^{92}(n,\gamma)$	γ	Nb X-RAYS
Aluminum	46Sc	0.229	$Sc^{45}(n,\gamma)$	$\beta^-,\beta^-,\gamma,\gamma$	1.48, 0.357, 0.889, 1.12
	54Mn	0.856	$Fe^{56}(d,\alpha)$	γ	0.835
	55Fe	2.8	$Fe^{54}(n,\gamma)$	γ	0.23*
	54Fe	0.122	$Fe^{52}(n,\gamma)$	β^-,γ,γ	1.57, 1.1, 1.29
	60Co	5.263	$Co^{59}(n,\gamma)$	β^-,γ,γ	0.314, 1.17, 1.33
	65Zn	0.667	$Zn^{64}(n,\gamma)$	γ,ϵ,β^+	1.115, 1.352, 0.325
	110mAg	0.69	$Ag^{109}(n,\gamma)$	β^-,γ^{**}	087, 6577

* Continuous spectrum of X-ray energies below this number, due to Bremmstrahlung.

** Energy of most probable energy β- and most probable energy γ given.

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Since preparing estimates of D&D wastes that would result from each of the 155 commercial reactors that will be operating in the country by the year 1994 is beyond the scope of this report, the data reported here are from published studies on reference LWRs.[7,8] These data are provided to give a rough indication of the waste volumes and radioactivity present in major neutron irradiated components in typical BWRs and PWRs in use in the United States. The actual D&D waste inventory that might result from the D&D of any particular reactor could vary significantly from the data presented here depending upon its size, operating history, and materials of construction.

Figures 4.3 and 4.4 show views of BWR and PWR reactor vessel and core components which are irradiated by neutrons during the operation of the reactor. These core components and the pressure vessel are major sources of radioactivity in D&D wastes (contain over 95% of all non-fuel radioactivity in a reactor). Tables 4.4 through 4.8 provide estimates of the radioactivity in BWR and PWR power plants at the end of their operating lives.[7,8] These estimates are from studies based on a reference 1155 MWe BWR[7] and a 1175 MWe PWR.[8] These reactors have been assumed to have been in operation for 30 effective full power years (EFPY). The material composition of the pressure vessel and core components are based on nominal handbook values for nuclear grade materials.

4.3.2 Other Radioactive Wastes

Contributions to the total radioactivity in D&D wastes from deposits of activated corrosion products in pipings and filters is a very small fraction of the total radioactivity from neutron activated components (less than 5%). Similarly, contaminated wastes from other parts of the reactor, though large in volume, contain negligible amounts of radioactivity. Most of these contaminated wastes (which have low specific activity) are expected to be disposed of in shallow land burials. These "other radioactive wastes" from nuclear power plants are not discussed in this report.

4.4 D&D Radwaste Projections

A list of presently operating commercial power reactors in the United States, and those likely to be on line by the year 1994 is given in Appendix A.[20]

Data from Appendix A have been used to plot Figure 4.5, showing the cumulative number of reactors in operation during any particular year between the years 1960 and 1994. Figure 4.6 shows the projection for the number of power reactors that might be in some stage of decommissioning during the years between 2000 and 2040. This plot is based on the assumption that the operating life of all power reactors is 40 years; at the end of which the reactor is promptly decommissioned followed by complete dismantling which could take three to six years. The graph shows that there could be as many as 44 to 65 commercial power reactors in various stages of decommissioning at any one time. The lower bound, i.e. 44, is based on the assumption that it takes three years to completely dismantle a reactor, while the upper bound, i.e. 65, represents a dismantling time of six years. Graphs for other assumed dismantling times are also plotted in Figure 4.6.

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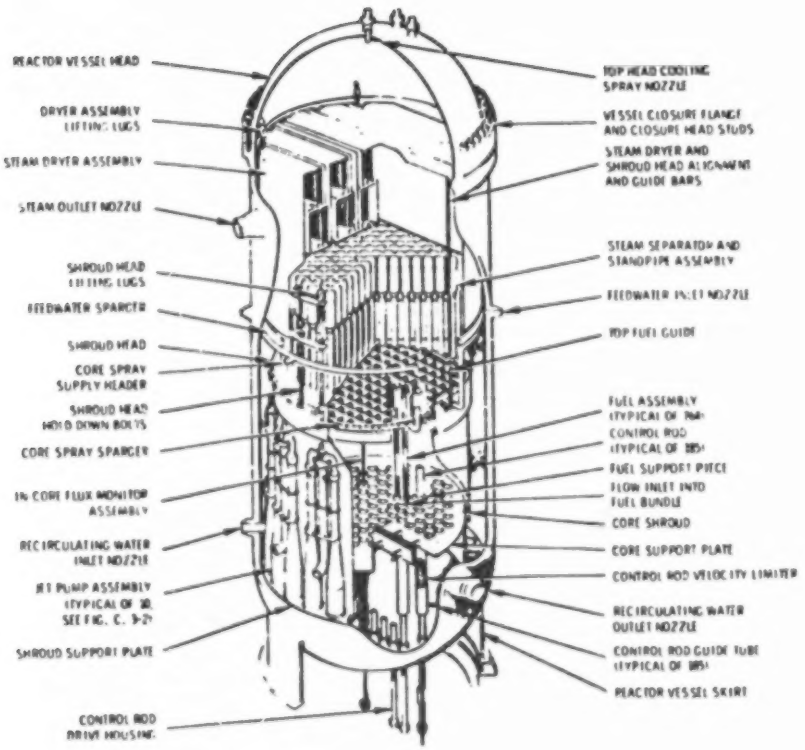
REACTOR VESSEL HEAD
DRYER ASSEMBLY LIFTING LUGS
STEAM DRYER ASSEMBLY
STEAM OUTLET NOZZLE
SHROUD HEAD LIFTING LUGS
FEEDWATER SPARGER
SHROUD HEAD
CORE SPRAY SUPPLY HEADER
SHROUD HEAD HOLD DOWN BOLTS
CORE SPRAY SPARGER
IN-CORE FLUX MONITOR ASSEMBLY
RECIRCULATING WATER INLET NOZZLE
JET PUMP ASSEMBLY (TYPICAL OF 10, SEE FIG. C. 3-2)
SHROUD SUPPORT PLATE
CONTROL ROD DRIVE HOUSING
TOP HEAD COOLING SPRAY NOZZLE
VESSEL CLOSURE FLANGE AND CLOSURE HEAD STUDS
STEAM DRYER AND SHROUD HEAD ALIGNMENT AND GUIDE BARS
STEAM SEPARATOR AND STANDPIPE ASSEMBLY
FEEDWATER INLET NOZZLE
TOP FUEL GUIDE
FUEL ASSEMBLY (TYPICAL OF 764)
CONTROL ROD (TYPICAL OF 185)
FUEL SUPPORT PIECE
FLOW INLET INTO FUEL BUNDLE
CORE SHROUD
CORE SUPPORT PLATE
CONTROL ROD VELOCITY LIMITER
RECIRCULATING WATER OUTLET NOZZLE
CONTROL ROD GUIDE TUBE (TYPICAL OF 185)
REACTOR VESSEL SKIRT

Figure 4.3. BWR pressure vessel and internals.[7]

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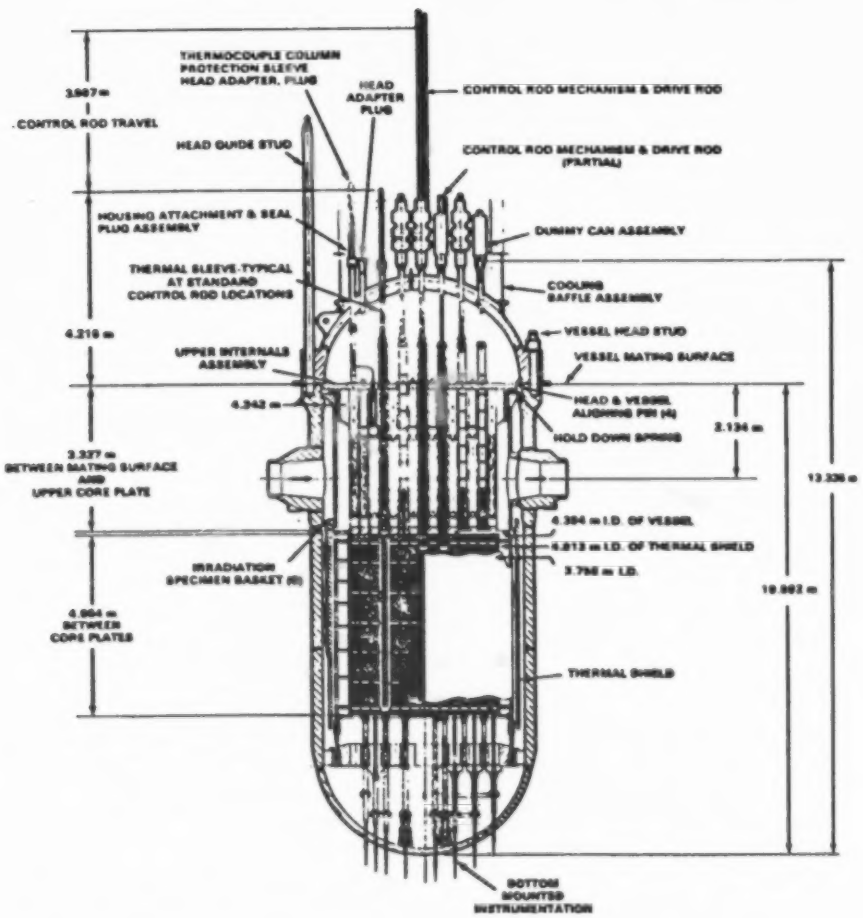
THERMOCOUPLE COLUMN PROTECTION SLEEVE HEAD ADAPTER, PLUG
HEAD ADAPTER PLUG
CONTROL ROD MECHANISM & DRIVE ROD
3.987 m
CONTROL ROD TRAVEL
HEAD GUIDE STUD
CONTROL ROD MECHANISM & DRIVE ROD (PARTIAL)
HOUSING ATTACHMENT & SEAL PLUG ASSEMBLY
DUMMY CAN ASSEMBLY
THERMAL SLEEVE-TYPICAL AT STANDARD CONTROL ROD LOCATIONS
COOLING BAFFLE ASSEMBLY
4.216 m
VESSEL HEAD STUD
UPPER INTERNALS ASSEMBLY
VESSEL MATING SURFACE
4.342 m
HEAD & VESSEL ALIGNING PIN (4)
2.134 m
HOLD DOWN SPRING
3.327 m BETWEEN MATING SURFACE AND UPPER CORE PLATE
13.336 m
4.394 m I.D. OF VESSEL
4.013 m I.D. OF THERMAL SHIELD
3.756 m I.D.
IRRADIATION SPECIMEN BASKET (6)
10.992 m
4.064 m BETWEEN CORE PLATES
THERMAL SHIELD
BOTTOM MOUNTED INSTRUMENTATION

Figure 4.4. PWR pressure vessel and internals.[8]

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Table 4.4

Reference Radionuclide Inventory 1,
Neutron Activated Stainless Steel[a] (BWR)
at Time of Reactor Shutdown[7]

Radionuclide	Radioactivity Concentration at Shutdown (Ci/m^3)	Fractional Radioactivity at Shutdown
^{3}H	4.65×10^{-4}	1.63×10^{-10}
^{10}Be	2.63×10^{-6}	---[b]
^{14}C	1.05×10^{2}	3.68×10^{-5}
^{32}P	1.11×10^{2}	3.89×10^{-5}
^{33}P	6.65×10^{2}	2.33×10^{-4}
^{35}S	5.52×10^{1}	1.94×10^{-5}
^{36}Cl	2.69×10^{-4}	---[b]
^{51}Cr	1.45×10^{6}	5.09×10^{-1}
^{54}Mn	8.50×10^{3}	2.98×10^{-3}
^{55}Fe	9.22×10^{5}	3.24×10^{-1}
^{59}Fe	2.74×10^{4}	9.61×10^{-3}
^{58}Co	2.10×10^{4}	7.37×10^{-3}
^{60}Co	3.36×10^{5}	1.18×10^{-1}
^{59}Ni	6.36×10^{2}	2.23×10^{-4}
^{63}Ni	8.75×10^{4}	3.07×10^{-2}
^{65}Zn	3.23×10^{1}	1.13×10^{-5}
^{93}Zr	8.15×10^{-6}	---[b]
^{95}Zr	1.41×10^{-2}	4.21×10^{-5}
^{93m}Nb	1.35×10^{-1}	4.74×10^{-8}
^{94}Nb	1.50×10^{0}	5.26×10^{-7}
^{95}Nb	1.20×10^{2}	4.21×10^{-5}
^{93}Mo	3.26×10^{-1}	1.14×10^{-7}
^{99}Tc	3.18×10^{-2}	1.12×10^{-8}
^{108m}Ag	7.36×10^{-2}	2.58×10^{-8}
^{108}Ag	8.67×10^{1}	3.04×10^{-5}
^{109m}Ag	3.61×10^{0}	1.27×10^{-6}
^{109}Cd	3.42×10^{0}	1.20×10^{-6}
^{110m}Ag	8.04×10^{0}	2.82×10^{-6}
^{110}Ag	2.02×10^{2}	7.09×10^{-5}
^{151}Sm	2.12×10^{-2}	7.44×10^{-9}
^{152}Eu	1.12×10^{-3}	3.93×10^{-10}
^{154}Eu	3.12×10^{0}	1.09×10^{-6}
^{160}Tb	9.48×10^{-4}	3.33×10^{-10}
^{166m}Ho	7.84×10^{-4}	2.75×10^{-10}
TOTAL	2.85×10^{6}	1.00

[a] Calculated at the inner surface of the 304 stainless steel core shroud, at the axial midplane of the fuel zone, for 30 EFPY of operation.

[b] Indicates a value of less than 1.00×10^{-10}.

Table 4.5

Reference Radionuclide Inventory 2,
Neutron Activated Carbon Steel[a] (BWR)
at Time of Reactor Shutdown[7]

Radionuclide	Radioactivity Concentration at Shutdown (Ci/m^3)	Fractional Radioactivity at Shutdown
^{14}C	6.77×10^{-3}	1.84×10^{-5}
^{32}P	9.20×10^{-1}	2.51×10^{-3}
^{35}S	1.76×10^{-2}	4.80×10^{-5}
^{51}Cr	1.75×10^{0}	4.77×10^{-3}
^{54}Mn	1.06×10^{1}	2.89×10^{-2}
^{55}Fe	3.35×10^{2}	9.13×10^{-1}
^{59}Fe	9.44×10^{0}	2.57×10^{-2}
^{58}Co	1.49×10^{0}	4.06×10^{-3}
^{60}Co	6.49×10^{0}	1.77×10^{-2}
^{59}Ni	1.46×10^{-2}	3.98×10^{-5}
^{63}Ni	1.73×10^{0}	4.71×10^{-3}
^{65}Zn	1.88×10^{-6}	5.12×10^{-9}
^{98}Sr	5.24×10^{-13}	1.43×10^{-15}
^{91}Y	2.25×10^{-13}	6.13×10^{-16}
^{95}Zr	1.62×10^{-4}	4.41×10^{-7}
^{93m}Nb	3.90×10^{-4}	1.06×10^{-6}
^{94}Nb	8.30×10^{-7}	2.26×10^{-9}
^{95}Nb	3.76×10^{-4}	1.02×10^{-6}
^{93}Mo	9.39×10^{-4}	2.56×10^{-6}
^{99}Tc	2.64×10^{-4}	7.19×10^{-7}
TOTAL	3.67×10^{2}	1.00

[a] Calculated at the inner surface of the SA 533 carbon steel reactor vessel, at the axial midplane of the fuel zone, for 30 EFPY of operation.

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Table 4.6

Estimated Total Radioactivity in Neutron Activated Components (BWR)[7]

Component (quantity)	Estimated Activated Volume (m^3)	Radioactivity per Component (Ci)	Estimated Total Radioactivity (Ci)	Estimated mass, tonnes
Core Shroud (1)	3.75	6.30×10^6	6.30×10^6	30.0
Jet Pump Assembly (10)	0.076	2.00×10^3	2.00×10^4	.6
Reactor Vessel (1)				
Cladding	0.428	4.58×10^2	2.16×10^3	3.4
Shell Wall*	15.26	1.70×10^3		122.0
Sacrificial Shield (1)				
Inner Shell	2.19	1.03×10^2		17.5
Reinforced Concrete Region	73.30	3.47×10^0	1.66×10^2	--
Outer Shell	6.22	5.39×10^1		49.7
Steam Separator Assembly (1)				
Shroud Head Plate	0.841	8.65×10^3	9.60×10^3	6.7
Steam Separator Risers	0.376	9.52×10^2		3.0
Top Fuel Guide (1)	0.310	3.01×10^4	3.01×10^4	22.8
Orificed Fuel Support (193)	0.0036	3.63×10^0	7.01×10^2	
Core Support Plate (1)	2.54	6.50×10^2	6.50×10^2	
Incore Instrument Strings (55)	0.00026	1.99×10^2	1.10×10^4	.04
Control Rod (185)	0.0019	9.61×10^2	1.78×10^5	
Control Rod Guide Tube (185)	0.0024	5.12×10^{-1}	9.47×10^1	
TOTAL			6.55×10^6	258.0

*Total mass of vessel including top head, studs, nuts, and washers is 748 tonnes.

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Table 4.7

Summary of Estimated Radioactivity Levels in Activated Steel Components (PWR)[8]

Component	Estimated Radioactivity (Ci)	Mass, tonnes
Pressure Vessel, Cylindrical Wall	19,171	269.8
Vessel Head	<10	88.45
Vessel Bottom	<10	38.9
Upper Core Upper Support Columns	<100	11.11
Upper Core Barrel	<1000	
Upper Core Grid Plate	24,310	4.63
Guide Tubes	<100	15.10
Lower Core Barrel	651,600	45.36[a]
Thermal Shields	146,100	10.43
Core Shroud	3,431,000	12.3
Lower Core Grid Plate	553,400	3.95
Lower Support Columns	<10,000	3.36
Lower Core Forging	<2,500	36.47
Misc. Internals	<2,000	36.29
Reactor Cavity Liner	<10	
TOTAL	<4,841,321	588.13

[a] Includes upper core barrel mass.

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Table 4.8

Radioactivity Levels in Major Activated Reactor Components at Time of Reactor Shutdown (PWR)[8]

Isotope	Half-Life	Core Mid-Plane Radioactivities (Ci/m^3)						
		Shroud	Lower 4.72 m of Core Barrel	Thermal Shields	Vessel Inner Cladding	Lower 5.02 m of Vessel Wall	Upper Grid Plate[a]	Lower Grid Plate[a]
^{95}Nb	35 day	2.0×10^{3}	7.6×10^{0}	3.5×10^{0}	5.6×10^{-3}	1.7×10^{-3}		
^{59}Fe	45 day	4.6×10^{4}	4.4×10^{3}	2.0×10^{3}	1.0×10^{2}	2.7×10^{1}		
^{58}Co	72 day	1.5×10^{5}	1.0×10^{4}	4.6×10^{3}	3.3×10^{2}	6.6×10^{0}		
^{95}Zr	65 day	1.1×10^{-1}	6.2×10^{-3}	2.9×10^{-3}	2.0×10^{-4}	7.2×10^{-4}		
^{65}Zn	245 day	1.2×10^{2}	1.1×10^{0}	5.0×10^{-1}	6.7×10^{-4}	3.5×10^{-5}		
^{54}Mn	~300 day	6.8×10^{4}	3.7×10^{3}	1.7×10^{3}	1.2×10^{2}	4.7×10^{1}		
^{55}Fe	2.7 yr	1.3×10^{6}	1.5×10^{5}	6.7×10^{4}	3.5×10^{3}	7.2×10^{2}		
^{60}Co[b] upper	5.27 yr	9.6×10^{5}	9.3×10^{4}	4.7×10^{4}	2.6×10^{3}	7.6×10^{1}		
lower		3.2×10^{5}	3.1×10^{4}	1.6×10^{4}	8.2×10^{2}	2.5×10^{1}		
^{63}Ni	~100 yr	1.2×10^{5}	1.8×10^{4}	6.8×10^{3}	3.6×10^{2}	3.8×10^{0}		
^{93}Mo	~3000 yr	3.6×10^{-1}	5.2×10^{-2}	2.4×10^{-2}	1.2×10^{-3}	1.3×10^{-3}		
^{14}C	~5,750 yr	1.5×10^{2}	1.8×10^{1}	8.3×10^{0}	4.0×10^{-1}	1.9×10^{-2}		
^{94}Nb	~20,000 yr	5.4×10^{0}	2.6×10^{-1}	1.2×10^{-1}	9.5×10^{-3}	--		
^{59}Ni	~80,000 yr	7.4×10^{2}	1.3×10^{2}	6.0×10^{1}	3.0×10^{0}	3.2×10^{-2}		
Sum (Ci/m^3)		2.97×10^{6}	3.87×10^{5}	1.45×10^{5}	7.73×10^{3}	9.04×10^{2}	2.97×10^{6}	2.97×10^{6}
Average/Peak		0.755	0.637	0.778	0.637	0.637	0.003 x 4.74[c]	0.08 x 4.74[c]
Ci/kg[d]		2.787×10^{2}	2.433×10^{1}	1.403×10^{1}	7.164×10^{-1}	7.164×10^{-2}	5.254×10^{0}	1.403×10^{2}
Weight of Material (kg)		12,312	26,783	10,413	2,074	245,582	4,627	3,946
Sum (Ci)[e]		3.431×10^{6}	6.516×10^{5}	1.461×10^{5}	1.581×10^{3}	1.769×10^{4}	2.431×10^{4}	5.534×10^{5}

Total - Radioactivity: 4.826×10^{6} Curies, 1.786×10^{17} becquerels

(a) Normalized to shroud

(b) Upper and lower bounds were computed using the maximum and minimum levels of ^{59}Co contaminant in the materials. All totals were computed using the upper bound values.

(c) $\frac{\text{Activity (Plate Average)}}{\text{Activity (Plate Edge)}} = 4.74$, $\frac{\text{Activity (Shroud at Plate Location)}}{\text{Activity (Shroud at Axial Midplane)}} = \begin{cases} 0.005, \text{ upper plate} \\ 0.08, \text{ lower plate} \end{cases}$

(d) Conversion factor assumes stainless steel density of 8.038×10^{3} kg/m^3 (0.29 $lb/in.^3$).

(e) The number of significant figures carried is for computational accuracy and does not imply precision to four places.

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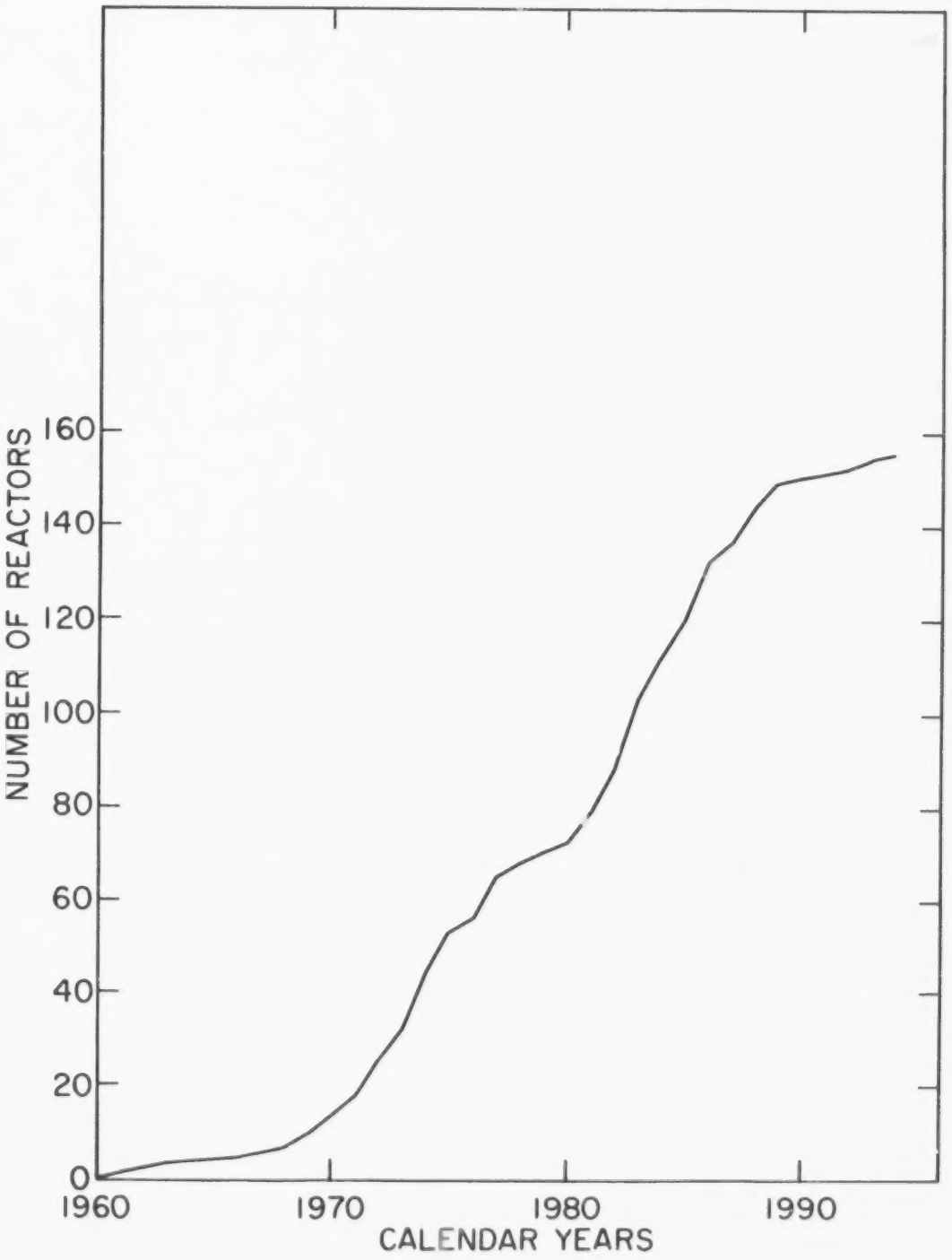

Figure 4.5. Projection of the cumulative number of commercial power reactors in the United States.

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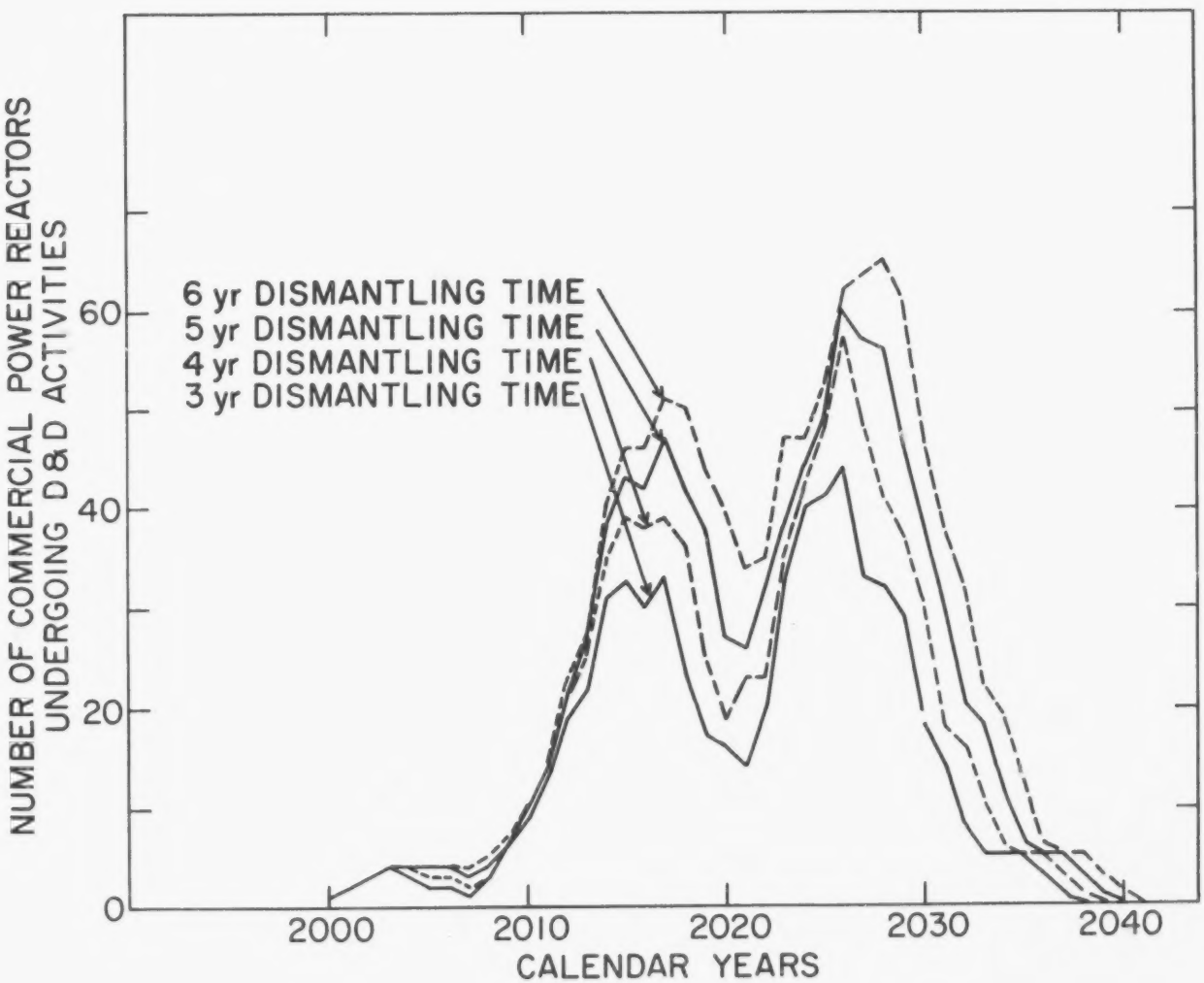

Figure 4.6. Projection of the number of commercial power reactors undergoing D&D activities.

How realistic these numbers turn out to be will depend upon the D&D program plans for the various power reactors (which are not known at present).

Since no commercial power reactor of any significant size has been decommissioned and dismantled completely as yet, the projections of D&D wastes from 150+ power reactors can only be based on the waste estimates from studies of reference facilities.[7,8] Some assumptions regarding scaling factors for wastes from reactors smaller than the reference LWRs have to be made. For the sake of discussion and obtaining some idea about the quantities of D&D wastes, it is assumed that the power reactors can be divided into four size ranges depending only upon their power output, viz. 30 to 300 MWe, 301 to 600 MWe, 601 to 900 MWe, and 901 to 1200 MWe. Further, an LWR of the size between 901 and 1200 MWe is assumed to generate neutron activated D&D wastes equal to estimates from reference reactors - which in the case of a BWR are 6.55×10^6 curies contained in 258 tonnes of activated materials (Table 4.6),[7] and in the case of a PWR are 4.84×10^6 curies contained in 588 tonnes of activated materials (Table 4.8).[8] For smaller size reactors, the above neutron activated D&D waste estimates are multiplied by a scaling factor: α_1 for 601 to 900 MWe, α_2 for 301 to 600 MWe, and α_3 for 30 to 300 MWe range reactors. The values of α_1, α_2, and α_3 are chosen here as 0.75, 0.50, and 0.25 for providing only rough estimates.

Based on the data for the commercial power reactors given in Appendix A and the assumptions described above, Figure 4.7 shows a plot of the mass of the neutron activated wastes that would be generated from all LWRs. Estimated waste quantities are approximately 1.1×10^4 tonnes from all BWRs by the year 2029, and approximately 5.1×10^4 tonnes from all PWRs by the year 2034. Cumulative radwaste mass (in tonnes) is also shown in the same figure. An accurate inventory of radioactivity in these components is much more difficult to make, since it is a function of material composition, reactor operating history, and the decay time before the reactor is dismantled. Rough estimates using scaling factors described earlier result in histograms shown in Figure 4.8. These histograms show the estimated radioactivity contribution (in curies) of the neutron activated D&D components at the time of reactor shutdown. To arrive at cumulative radioactivity projections, decay corrections will have to be applied to each of the radionuclide species present in the D&D waste.

At this time, these rough projections of D&D radwaste from commercial power reactors do not lend themselves to any detailed interpretation or discussion.

5. PARTICLE ACCELERATORS

There are perhaps as many as 1,200 particle accelerators currently operating in the United States, ranging in size from very small Cockroft-Walton and electron linear accelerators to multi-GeV research synchrotrons. Of these, at least 50 accelerators produce fluxes that cause significant induced activation, and several hundred more are capable of producing neutron fluxes that could result in non-negligible activation of various components of the accelerator facility. Almost all accelerators producing non-negligible activation are owned or funded by an agency of the federal government. Appendix B1 provides a list of

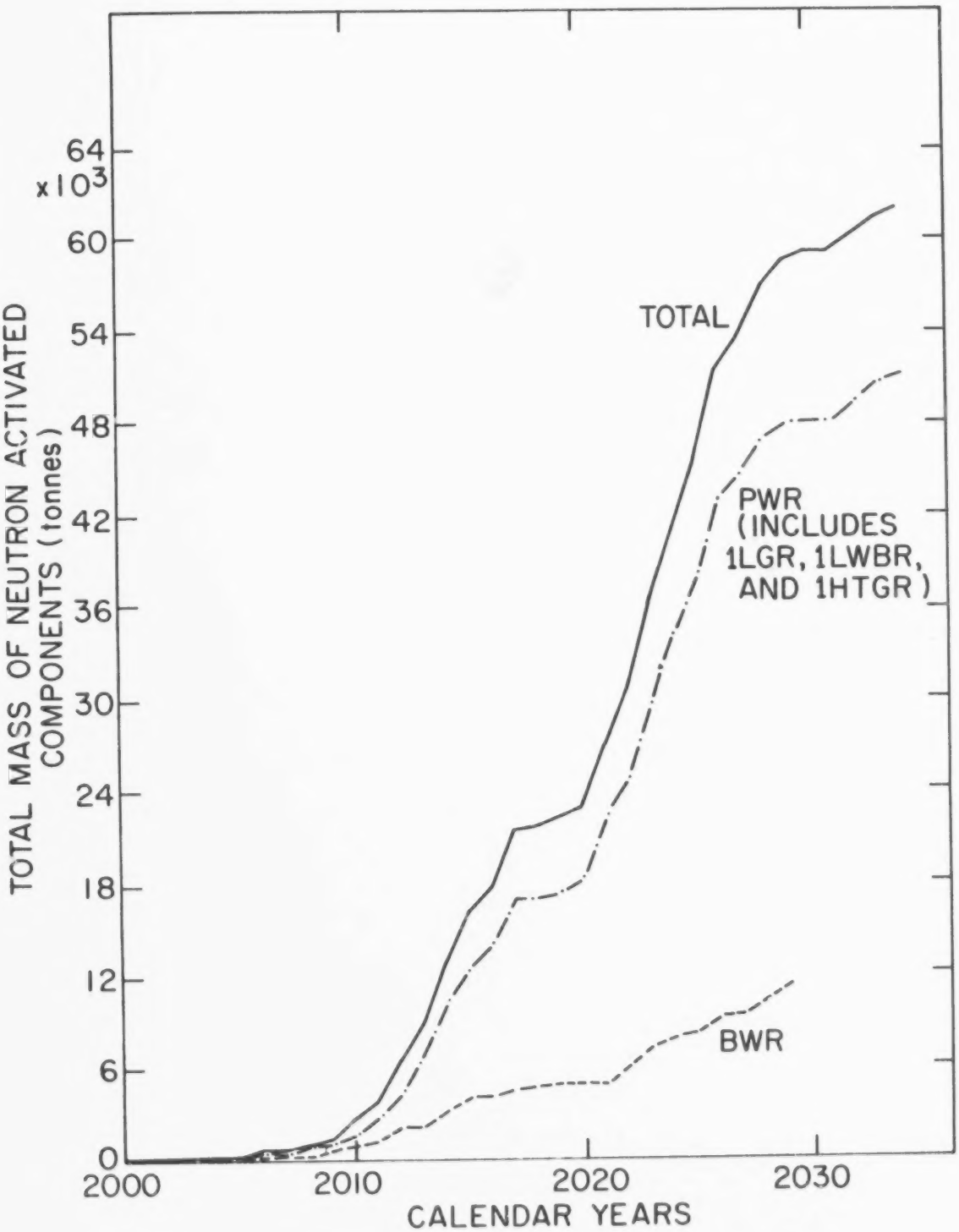

Figure 4.7. Projection of the cumulative mass of neutron activated D&D wastes from commercial power reactors.

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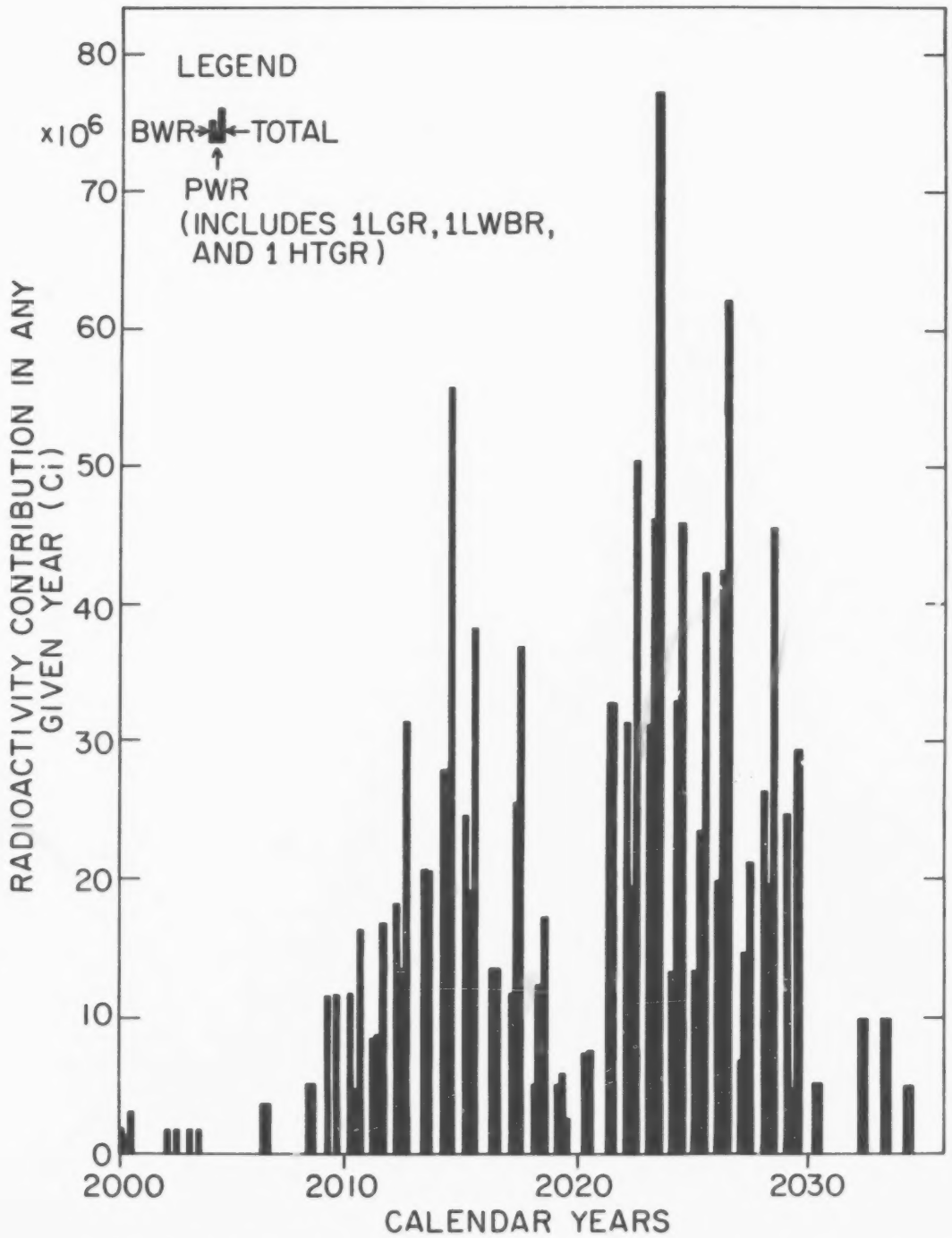

Figure 4.8. Projection of radioactivity inventory from decommissioned commercial power reactors.

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such accelerators. Appendix B2 provides a list of the accelerators that have already been decommissioned.

5.1 Types of Accelerators

Since the particle accelerators vary widely in terms of size, energy and design, it is necessary to deal with the estimates of D&D radwaste in a generic way. In the discussion that follows, the accelerators have been divided into the following categories:

- Positive-ion electrostatic accelerators.
- Electron accelerators.
- Cyclotrons and synchrocyclotrons.
- Proton synchrotrons and linacs.

5.2 Sources of D&D Radwaste

In general, all accelerators, positive-ion or electron with energies >10 MeV will produce induced activity in certain parts of the machine.[10] In most cases the induced radioactivity is confined to relatively few parts of the accelerator, viz. direct beam components, beam surroundings, and the building. Table 5.9 lists the likelihood of radioactivity being present in these three categories of components in various types of accelerators.

5.3 Radionuclides of Concern

The major induced radioactivity is confined to the target material and the beam stop. The beam stop is generally a block of concrete or iron. Various materials could be used as a target material. Some of the commonly used materials are listed in Table 5.10. The radionuclides produced as a result of bombardment of the target and the beam stop with energetic particles include Na-22 in concrete, Na-22 and Al-26 in aluminum, Mn-54 in iron, Co-57 in nickel, and Co-60 and Ni-63 in copper.[10]

5.4 Radioactive Waste Inventory

The inventory of the induced radioactivity in a particle accelerator at the time of D&D can be calculated by the procedures reported in the literature.[11-13] As an example, Table 5.10 provides a summary of the long-lived isotope-production data calculated for electron beams with energies of 35 MeV and greater. The percentage saturation obtained for each isotope after an assumed 25-year operating life of the machine has been calculated, and is shown in the table along with the expected production in terms of curies per kilowatt of beam power.[10] In the case of a high energy proton accelerator, the number of curies per microampere of beam current is given by (170 x Ep), where Ep is the proton energy in GeV.[10] Hence, for a beam current of 0.16 uA, a 400-GeV beam will produce 10 kCi of activity at equilibrium, which is obtained after long operating times.[10]

Table 5.11 summarizes the quantities of radioactive materials from a few selected accelerators, in terms of mass and curie content, that will require proper disposal upon decommissioning and dismantling of these machines. A

Table 5.9

Radioactivity Expected in Three Categories of Components From Various Types of Accelerators[10]

Accelerator Type	Energy	Categories of Radioactive Waste		
		Direct-Beam Components	Beam Surroundings	Bldg.
Positive-ion electrostatic accelerators	< 1 MeV	None	None	None
Electron accelerators	< 10 MeV	None	None	None
Positive-ion electrostatic accelerators	> 1 MeV	a	b	None
Electron accelerators	> 10 MeV	a	b	None
	<100 MeV	a	b	None
Cyclotrons, synchrocyclotrons, and electron accelerators	>100 MeV	a	b	b
All proton synchrotrons and LAMPF		a	a	a

[a]Definitely radioactive.
[b]Possibly radioactive.

Table 5.10

Induced Radioactivity From Electron Beams[10]

Target Material	Isotope Produced	Half Life	Saturation Activity[a] (Ci/kW)	Percent Saturation for 25-Year Operation	Actual Expected Activity (Ci/kW)
Concrete	^{22}Na	2.6 yr	0.1	99.9	0.10
Aluminum	^{22}Na	2.6 yr	0.28	99.9	0.28
	^{26}Al	7.4×10^5 yr	8.8	0.0023	2.04×10^{-4}
Iron	^{54}Mn	303 d	0.59	100.0	0.59
Nickel	^{57}Co	270 d	5.9	100.0	5.9
	^{60}Co	5.26 yr	0.1	96.2	0.096
Copper	^{60}Co	5.26 yr	0.65	96.2	0.063
	^{63}Ni	92 yr	0.45	17.0	0.077

[a]From "Radiological Safety Aspects of the Operation of Electron Linear Accelerators," W. P. Swanson (SLAC), IAEA, Vienna, STI/DOC/10/, ISBN 92-0-00000-0, 1978.

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Table 5.11

Mass of Radioactive Waste From Selected Accelerators[10]

Location	Energy	Diameter or Length	Funding Source	Estimated Mass of Material [> 1 R/h @ 1 m] (kg)	Reported Mass of Material [> 2.5 mR/h @ 1 m] (kg)	Reported Mass of Accelerator (kg)	Total Mass of Active Material (kg)	Estimated Amount of Radioactivity (Ci)	Time-Averaged Beam Current (μA)	Primary Usage(s) of Accelerator[a]	Remarks
Positive-ion cyclotrons and synchrocyclotrons											
Duke U	15 MeV	31"	DOE/NSF/HEW	--	--	--	--	7.6 E+01	30	r	
Medi-Phy, Emeryville, CA	20 MeV	38"	Private	--	--	2.0 E+04	2.2 E+04	1.5 E+03	400	i	
Medi-Phy, So. Plfld, NJ	22 MeV	38"	Private	--	--	2.0 E+04	2.2 E+04	1.5 E+03	400	i	
New England Nuclear	22 MeV	38"	Private	--	--	2.0 E+04	2.2 E+04	4.1 E+02	110	i,d	
UCLA	22 MeV	38"	DOE/UCLA	--	2.3 E+03	2.3 E+04	2.5 E+04	1.1 E+02	30	i	
U of Colorado	28 MeV	52"	DOE/State	--	2.8 E+03	9.0 E+04	9.9 E+04	7.6 E+02	160		
ANL	40 MeV	60"	DOE	4.8 E+01	8.7 E+03	2.6 E+05	3.2 E+05	--	--	r,i	
U of Washington (Seattle)	40 MeV	60"	--	1.1 E+02	2.7 E+04	1.1 E+05	1.2 E+05	--	--	r	
Michigan State U	50 MeV	67"	NSF/MSU	2.5 E+00	3.9 E+02	2.0 E+05	2.2 E+05	1.7 E+02	20	r	
NASA/Lewis	55 MeV	83"	NASA	5.4 E+01	2.3 E+05	2.3 E+05	2.3 E+05	2.3 E+02	25	r	
LBL	60 MeV	88"	DOE	1.4 E+01	6.7 E+02	3.0 E+05	3.3 E+05	1.9 E+03	190	r	
Texas A&M	60 MeV	88"	DOE/State/ Private	0	6.4 E+04	2.7 E+05	3.0 E+05	2.9 E+03	285	r	
Harvard U	160 MeV	95"	NSF	4.2 E+02	8.0 E+03	7.1 E+05	7.8 E+05	5.4 E+02	0.2	m,r	
LBL	730 MeV	184"	DOE	--	--	4.3 E+06	4.7 E+06	8.2 E+01	0.5	m	
NASA/SREL	600 MeV	197"	NASA/NSF/ State	2.0 E+04	1.2 E+05	2.6 E+06	2.9 E+06	2.0 E+05	2.0	r	Shutdown Aug 78
Indiana U	225 MeV	272"	NSF/IU	--	--	2.0 E+06	2.2 E+06	--	--	r	
Proton synchrotrons											
Princeton-Penn	3 GeV	60'	DOE	--	--	1.4 E+06	1.5 E+06[b]	5.1 E+01	0.1	r	Shutdown 8 Apr 72
LBL-Bevatron	6.2 GeV	100'	DOE	--	--	9.3 E+06	1.1 E+07	9.5 E+02	0.9	r	w/50-MeV linac
ANL-ZGS	12.7 GeV	180'	DOE	2.3 E+04	9.5 E+06	1.4 E+07	1.5 E+07	2.2 E+02	0.1	r	w/50-MeV linac
BNL-AGS	33 GeV	840'	DOE	6.0 E+06	3.7 E+07	1.8 E+08	2.0 E+08[b]	2.2 E+03	0.4	r	w/200-MeV linac
FNAL	400 GeV	5000'	DOE	4.6 E+05	2.0 E+06	1.0 E+07	1.1 E+07[c]	6.8 E+03	0.1	r	w/8-GeV booster synchrotron and 200-MeV linac
Electron synchrotron											
Cornell U	12 GeV	820'	NSF	--	--	1.3 E+05	--	--	0.8	r	
Electron linacs											
IRT San Diego	35 MeV	30'	Private	0	3.1 E+02	1.0 E+04 to 1.0 E+05	--	--	500		
ANL	35 MeV	30'	DOE	0	4.0 E+01	4.0 E+04[d]	--	--	--	r	Includes power supplies, etc.
MIT Bates Lab	400 MeV	600'	DOE	--	5.5 E+04[d]	--	--	--	--	r	
SLAC	20 GeV	2 mi	DOE	--	3.6 E+05	--	--	--	30	r	
Proton linacs											
LAMPF	800 MeV	2600'	DOE	--	5.5 E+07	--	--	9.5 E+04	700[e]	r	
Electrostatic accelerators											
ANL-Tandem	30 MeV		DOE	0	--	5.0 E+04	2.5 E+02	--	--		Heavy-ion work
U of Washington Tandem	--		--	0	--	5.0 E+04	3.0 E+00	--	--		

[a] Code: r = research, i = isotope production, d = machine development, and m = medical.

[b] Total includes some shielding.

[c] No shield material.

[d] Accelerator tank and beam stops only.

[e] Design goal (120 h/wk @ 1000 μA).

definite time table as to when each of the existing 1,200+ accelerators will be considered as candidates for D&D is not known at the present.

Figure 5.9, from another published study on accelerator decommissioning,[14] shows a plot of the estimated amount of material that has become radioactive at selected accelerators. The data plotted here are divided into three main categories: (a) medical and industrial accelerators, (b) research cyclotrons, and (c) the large research machines, which are shown individually. The total annual non-fuel cycle radwastes presently generated in the country are shown for comparison.[14]

5.5 Discussion

Although the mass of the radioactive material that would potentially result from the D&D of particle accelerators is quite large, the total radionuclide inventory (in terms of curie content) is quite small in comparison with the induced radioactivity in neutron activated materials in power reactors. For example, the data listed in Table 5.11 show the largest amount of D&D radionuclide inventory from any single machine (NASA/SREL accelerator) as 2×10^5 curies compared to 4.8 to 5.5×10^6 curies from a typical LWR in the 1155 to 1175 MWe size range.[7,8]

Also, it should be kept in mind that most of the radioactivity in an accelerator is confined to a very few components, viz. the target material, the magnet, the beam stop, and shielding. Unlike the neutron activated materials in a power reactor which cannot be reused because of the corrosion and structural material property degradation due to neutron irradiation, the major parts of an accelerator can be salvaged upon decommissioning and reused for building another machine elsewhere. This has traditionally been the case in the past and has resulted in reduction of the actual high specific activity D&D wastes that had to be disposed of to insignificant amounts.

6. CONCLUSIONS

This brief report points out the major sources of high activity wastes that would result from the decommissioning and dismantling (D&D) of power reactors, particle accelerators, and surplus DOE nuclear facilities. From the waste management point of view, the neutron activated reactor components, viz. the pressure vessel and the core structural components are of greatest concern. These components contain almost all of the radioactivity of the defueled nuclear reactor at shut down. Radionuclides of principal concern are Fe-55, Co-60, Ni-63, and C-14.

Projections of the number of reactors that might be in some stage of decommissioning and dismantling have been made for the years 2000 through 2040, when the major commercial power reactor D&D activities are expected to take place. Depending upon the time taken to completely dismantle a reactor, there could be as many as 44 to 65 power reactors in various stages of decommissioning in a particular year.

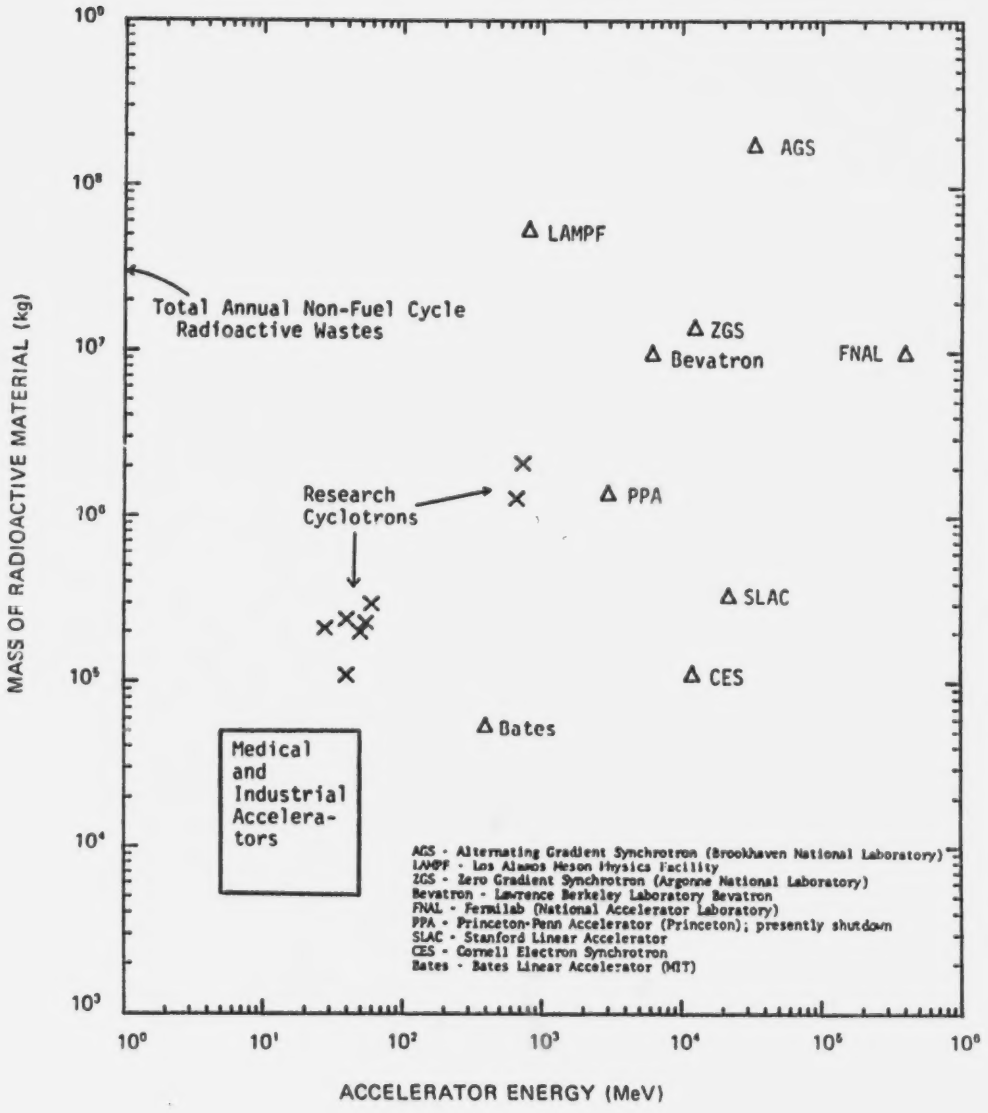

Figure 5.9. Induced activity in selected accelerators.[14]

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Estimates of the quantity of high activity D&D waste expected from the 155 commercial power plants (presently operating or expected to be on-line by 1994) show that the total mass of neutron activated components is approximately 6.3×10^4 tonnes. This amount does not include the mass of the pipings with deposited radioactive corrosion products. Since no commercial size power plant has been completely dismantled to date, the accuracy of any estimates of D&D wastes from power reactors is as yet unverifiable. The waste inventory projections given in this report are only rough estimates made under the assumptions stated.

From the survey of the published literature on the decommissioning and dismantling (D&D) of nuclear facilities (other than commercial power reactors), it is concluded that not enough information exists at this time to prepare a detailed inventory projection through the year 2000. A substantial number of additional studies are warranted to provide the necessary data in terms of mass, volume, and isotopic makeup for a complete inventory. However, waste projections based only on D&D waste volumes from surplus DOE facilities show a cummulative total of approximately 9.5×10^3 m^3 of TRU and approximately 1.1×10^5 m^3 of non-TRU wastes by the year 2000. Most of these wastes are low specific activity wastes.

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APPENDIX A

List of Commercial Power Reactors in the U. S. Presently Operating or Expected to be On-Line by 1994[20]

nuclear news

World List of Nuclear Power Plants

Operable, Under Construction, or on Order (30 MWe and Over) as of December 31, 1980

Twice each year Nuclear News sends a questionnaire to each utility or agency on this list, asking for corrections or additions to the information listed. In cases where a response is not received, we do follow up by phone, though such follow-up is not always possible for plants outside the United States.

The criterion for listing a unit is that either an order or a letter of intent has been signed for the reactor. In cases where the definition of "letter of intent" may be ambiguous, or where a special situation may exist, the judgment of the utility is followed as to whether a plant should be included in the list.

	Net MWe	Type	Reactor Supplier	Generator Supplier	Architect Engineer	Constructor	Construction stage (%)	Commercial Operation orig. sched-ule†	Commercial Operation actual or expected
United States									
NORTHEAST									
Baltimore Gas & Electric Co.									
• Calvert Cliffs 1 (Lusby, Md.)	850	PWR	C-E	GE	Bechtel	Bechtel	100	1/73	5/75
• Calvert Cliffs 2 (Lusby, Md.)	850	PWR	C-E	W	Bechtel	Bechtel	100	1/74	4/77
Boston Edison Co.									
• Pilgrim 1 (Plymouth, Mass.)	670	BWR	GE	GE	Bechtel	Bechtel	100	10/71	12/72
Pilgrim 2 (Plymouth, Mass.)	1150	PWR	C-E	GE	Bechtel	Bechtel	0	8/80	indef.
Connecticut Yankee Atomic Power Co.									
• Haddam Neck (Haddam Neck, Conn.)	575	PWR	W	W	S&W	S&W	100	11/67	1/68

• Units in commercial operation

† Estimated date of startup, announced at time reactor was ordered

cp — construction permit issued

lwa — limited work authorization issued

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	Net MWe	Type	Reactor Supplier	Generator Supplier	Architect Engineer	Constructor	Construction stage (%)	Commercial Operation orig. sched- uled	Commercial Operation actual or ex- pected
U.S.—NORTHEAST, cont'd									
Consolidated Edison Co.									
• Indian Point 2 (Indian Point, N.Y.)	873	PWR	W	W	UE&C	Wedco	100	6/69	7/74
Duquesne Light Co.									
• Beaver Valley 1 (Shippingport, Pa.)	833	PWR	W	W	S&W	S&W	100	6/73	4/77
Beaver Valley 2 (Shippingport, Pa.)	833	PWR	W	W	S&W	DLC/S&W	41	10/78	5/86
Duquesne Light Co. and U.S. Department of Energy									
• Shippingport (Shippingport, Pa.)	60	LWBR	W	W	S&L		100		12/77
Jersey Central P&L Co.									
• Oyster Creek 1 (Forked River, N.J.)	620	BWR	GE	GE	B&R/GE	B&R	100	2/68	12/69
Long Island Lighting Co.									
Shoreham (Brookhaven, N.Y.)	820	BWR	GE	GE	S&W	Utility	85	/75	1/83
Maine Yankee Atomic Power Co.									
• Maine Yankee (Wiscasset, Me.)	790	PWR	C-E	W	S&W	S&W	100		12/72
Metropolitan Edison Co.									
• Three Mile Island 1 (Londonderry Twp., Pa.)	792	PWR	B&W	GE	Gilbert	UE&C	100	9/71	9/74
• Three Mile Island 2 (Londonderry Twp., Pa.)	880	PWR	B&W	W	B&R	UE&C	100	5/73	12/78
Niagara Mohawk Power Corp.									
• Nine Mile Point 1 (Scriba, N.Y.)	610	BWR	GE	GE	Utility	S&W	100	11/68	12/69
Nine Mile Point 2 (Scriba, N.Y.)	1080	BWR	GE	GE	S&W	S&W	38	7/78	10/86
Northeast Utilities									
• Millstone 1 (Waterford, Conn.)	652	BWR	GE	GE	Ebasco	Ebasco	100	6/69	12/70
• Millstone 2 (Waterford, Conn.)	870	PWR	C-E	GE	Bechtel	Bechtel	100	4/74	12/75
Millstone 3 (Waterford, Conn.)	1150	PWR	W	GE	S&W	S&W	36	3/78	5/86
Pennsylvania Power & Light Co.									
Susquehanna 1 (Berwick, Pa.)	1050	BWR	GE	GE	Bechtel	Bechtel	87	5/79	4/82
Susquehanna 2 (Berwick, Pa.)	1050	BWR	GE	GE	Bechtel	Bechtel	55	5/81	4/83
Philadelphia Electric Co.									
• Peach Bottom 2 (Peach Bottom, Pa.)	1065	BWR	GE	GE	Bechtel	Bechtel	100	/71	7/74
• Peach Bottom 3 (Peach Bottom, Pa.)	1065	BWR	GE	GE	Bechtel	Bechtel	100	/73	12/74
Limerick 1 (Pottstown, Pa.)	1055	BWR	GE	GE	Bechtel	Bechtel	56	8/78	4/85
Limerick 2 (Pottstown, Pa.)	1055	BWR	GE	GE	Bechtel	Bechtel	18	1/80	4/87
Power Authority of the State of New York									
• Indian Point 3 (Indian Point, N.Y.)	965	PWR	W	W	UE&C	Wedco	100	7/71	8/76
• James A. FitzPatrick (Scriba, N.Y.)	821	BWR	GE	GE	S&W	S&W	100	1/73	7/75
Public Service Co. of New Hampshire									
Seabrook 1 (Seabrook, N.H.)	1150	PWR	W	GE	UE&C	UE&C	43.3	11/79	12/83
Seabrook 2 (Seabrook, N.H.)	1150	PWR	W	GE	UE&C	UE&C	7.7	8/81	/85
Public Service Electric & Gas Co.									
• Salem 1 (Salem, N.J.)	1090	PWR	W	W	Utility	UE&C	100	/71	6/77
Salem 2 (Salem, N.J.)	1115	PWR	W	W	Utility	UE&C	100	/73	4/81
Hope Creek 1 (Salem, N.J.)	1070	BWR	GE	GE	Bechtel	Bechtel	25	3/75	12/86
Hope Creek 2 (Salem, N.J.)	1070	BWR	GE	GE	Bechtel	Bechtel		3/77	12/89
Rochester Gas & Electric Corp.									
• Robert E. Ginna (Ontario, N.Y.)	490	PWR	W	W	Gilbert	Bechtel	100	11/69	3/70
Vermont Yankee Nuclear Power Corp.									
• Vermont Yankee (Vernon, Vt.)	514	BWR	GE	GE	Ebasco	Ebasco	100	10/70	11/72
Yankee Atomic Electric Co.									
• Yankee (Rowe, Mass.)	175	PWR	W	W	S&W	S&W	100	1/61	6/61
MIDWEST									
Cincinnati Gas & Light Co.									
Zimmer 1 (Moscow, Ohio)	810	BWR	GE	W	S&L	Kaiser	94	/75	/82
The Cleveland Electric Illuminating Co.									
Perry 1 (North Perry, Ohio)	1205	BWR	GE	GE	Gilbert	Kaiser	62.9	7/79	5/84
Perry 2 (North Perry, Ohio)	1205	BWR	GE	GE	Gilbert	Kaiser	39.5	7/80	5/88

CONTINUED

NOTE: Forked River-1, NYSEG-1 and -2, Montague-1 and -2, and North Anna-4 have been removed from this list. These units have all been canceled. Retained on this list is Humboldt Bay-3, which was taken out of service in July 1976 for refueling, maintenance, seismic modification, and a study of the geology of the area to answer questions posed by the NRC staff. Although Pacific Gas and Electric's application for a license to restart Humboldt Bay-3 was withdrawn in December 1980, the utility is still considering the possibility of restarting the plant should it prove economical to do so (see p. 44, this issue).

234 BEST DOCUMENT AVAILABLE

World List of Nuclear Power Plants, cont'd

	Net MWe	Type	Reactor Supplier	Generator Supplier	Architect Engineer	Constructor	Construction stage (%)	Commercial Operation orig. sched- uled†	Commercial Operation actual or expected
U.S.—MIDWEST, cont'd									
Commonwealth Edison Company									
• Dresden 1 (Morris, Ill.)	207	BWR	GE	GE	Bechtel	Bechtel	100	7/60	8/60
• Dresden 2 (Morris, Ill.)	794	BWR	GE	GE	S&L	UE&C	100	2/69	8/70
• Dresden 3 (Morris, Ill.)	794	BWR	GE	GE	S&L	UE&C	100	2/70	10/71
LaSalle 1 (Seneca, Ill.)	1078	BWR	GE	GE	S&L	Utility	99	10/75	4/82
LaSalle 2 (Seneca, Ill.)	1078	BWR	GE	GE	S&L	Utility	80	10/76	12/82
• Zion 1 (Zion, Ill.)	1040	PWR	W	W	S&L	Utility	100	4/72	10/73
• Zion 2 (Zion, Ill.)	1040	PWR	W	W	S&L	Utility	100	5/73	9/74
Byron 1 (Byron, Ill.)	1120	PWR	W	W	S&L	Utility	72	10/78	10/83
Byron 2 (Byron, Ill.)	1120	PWR	W	W	S&L	Utility	58	10/79	10/84
Braidwood 1 (Braidwood, Ill.)	1120	PWR	W	W	S&L	Utility	59	10/79	10/85
Braidwood 2 (Braidwood, Ill.)	1120	PWR	W	W	S&L	Utility	46	10/80	10/86
Carroll County 1 (Savanna, Ill.)	1120	PWR	W		S&L		0	10/87	10/92
Carroll County 2 (Savanna, Ill.)	1120	PWR	W		S&L		0	10/88	10/93
Commonwealth Edison Co. and Iowa-Illinois Gas & Electric Co.									
• Quad-Cities 1 (Cordova, Ill.)	789	BWR	GE	GE	S&L	UE&C	100	3/70	8/72
• Quad-Cities 2 (Cordova, Ill.)	789	BWR	GE	GE	S&L	UE&C	100	3/71	10/72
Consumers Power Co.									
• Big Rock Point (Charlevoix, Mich.)	63	BWR	GE	GE	Bechtel	Bechtel	100	12/62	12/62
• Palisades (South Haven, Mich.)	740	PWR	C-E	W	Bechtel	Bechtel	100	7/70	12/71
Midland 1 (Midland, Mich.)	530[a]	PWR	B&W	GE	Bechtel	Bechtel	62	5/78	7/84
Midland 2 (Midland, Mich.)	805	PWR	B&W	GE	Bechtel	Bechtel	62	5/79	12/83
Dairyland Power Cooperative									
• La Crosse BWR (Genoa, Wis.)	50	BWR	Allis	Allis	S&L	Maxon	100	10/66	11/69
Detroit Edison Co.									
Fermi 2 (Newport, Mich.)	1100	BWR	GE	EE	Utility	Daniel	76	/74	11/83
Illinois Power Co.									
Clinton 1 (Clinton, Ill.)	950	BWR	GE	GE	S&L	Baldwin	75	6/80	8/83
Clinton 2 (Clinton, Ill.)	950	BWR	GE	GE	S&L	Baldwin	0-cp	6/83	indef.
Indiana & Michigan Electric Co.									
• Donald C. Cook 1 (Bridgman, Mich.)	1054	PWR	W	GE	AEPSC	AEPSC	100	4/72	8/75
• Donald C. Cook 2 (Bridgman, Mich.)	1094	PWR	W	BB	AEPSC	AEPSC	100	4/73	7/78
Iowa Electric Light & Power Co.									
• Duane Arnold (Palo, Iowa)	545	BWR	GE	GE	Bechtel	Bechtel	100	12/73	5/74
Iowa Power & Light Co.									
Central Iowa (Vandalia, Iowa)	1270	PWR	B&W		Bechtel		0		indef.
Kansas Gas & Electric Co. and Kansas City Power & Light Co.									
Wolf Creek (Burlington, Kans.)	1150	PWR	W	GE	Bech/S&L	Daniel	70	4/81	4/84
Nebraska Public Power District									
• Cooper (Brownville, Neb.)	778	BWR	GE	W	B&R	B&R	100	4/71	7/74
Northern Indiana Public Service Co.									
Bailly N-1 (Baillytown, Ind.)	645	BWR	GE	GE	S&L	Braun	1	3/76	6/89
Northern States Power Co.									
• Monticello (Monticello, Minn.)	536	BWR	GE	GE	Bechtel	Bechtel	100	5/70	7/71
• Prairie Island 1 (Red Wing, Minn.)	520	PWR	W	W	FPS	Utility	100	5/72	12/73
• Prairie Island 2 (Red Wing, Minn.)	520	PWR	W	W	FPS	Utility	100	5/74	12/74
Omaha Public Power District									
• Fort Calhoun 1 (Fort Calhoun, Neb.)	490	PWR	C-E	GE	G&H	G&H	100	6/71	9/73
Public Service Indiana									
Marble Hill 1 (Jefferson County, Ind.)	1130	PWR	W	W	S&L	Utility	23	/82	late 86
Marble Hill 2 (Jefferson County, Ind.)	1130	PWR	W	W	S&L	Utility	11	/84	late 87
Toledo Edison Co.									
• Davis-Besse 1 (Oak Harbor, Ohio)	906	PWR	B&W	GE	Bechtel	Bechtel	100	12/74	11/77
Union Electric Co.									
Callaway 1 (Fulton, Mo.)	1150	PWR	W	GE	Bechtel	Daniel	75	10/81	4/83
Callaway 2 (Fulton, Mo.)	1150	PWR	W	GE	Bechtel	Daniel	0.5	4/83	/88
Wisconsin Electric Power Co.									
• Point Beach 1 (Two Creeks, Wis.)	497	PWR	W	W	Bechtel	Bechtel	100	4/70	12/70
• Point Beach 2 (Two Creeks, Wis.)	497	PWR	W	W	Bechtel	Bechtel	100	4/71	10/72
Wisconsin Public Services Corporation									
• Kewaunee (Carlton, Wis.)	535	PWR	W	W	FPS	FPS	100	6/72	6/74

[a] Midland-1 and -2 have the same size reactors; part of Unit 1's steam will be used in a nearby chemical facility

• Units in commercial operation

† Estimated date of startup, announced at time reactor was ordered

cp — construction permit issued

lwa — limited work authorization issued

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SOUTH	Net MWe	Type	Reactor Supplier	Generator Supplier	Architect Engineer	Constructor	Construction stage (%)	Commercial Operation orig. sched- uled†	Commercial Operation actual or expected
Alabama Power Company									
• Joseph M. Farley 1 (Dothan, Ala.)	860	PWR	W	W	SS/Bechtel	Daniel	100	4/75	12/77
Joseph M. Farley 2 (Dothan, Ala.)	860	PWR	W	W	SS/Bechtel	Daniel	90	4/76	5/81
Arkansas Power & Light Co.									
• Nuclear One 1 (Russellville, Ark.)	836	PWR	B&W	W	Bechtel	Bechtel	100	7/72	12/74
• Nuclear One 2 (Russellville, Ark.)	858	PWR	C-E	GE	Bechtel	Bechtel	100	12/75	3/80
Carolina Power & Light Co.									
• Robinson 2 (Hartsville, S.C.)	665	PWR	W	W	Ebasco	Ebasco	100	5/70	3/71
• Brunswick 1 (Southport, N.C.)	790	BWR	GE	GE	UE&C	Brown	100	3/75	3/77
• Brunswick 2 (Southport, N.C.)	790	BWR	GE	GE	UE&C	Brown	100	3/74	11/75
Shearon Harris 1 (Newhill, N.C.)	900	PWR	W	W	Ebasco	Daniel	37	3/77	9/85
Shearon Harris 2 (Newhill, N.C.)	900	PWR	W	W	Ebasco	Daniel	3	3/78	3/88
Shearon Harris 3 (Newhill, N.C.)	900	PWR	W	W	Ebasco	Daniel	<1	3/79	3/94
Shearon Harris 4 (Newhill, N.C.)	900	PWR	W	W	Ebasco	Daniel	<1	3/80	3/92
Duke Power Co.									
• Oconee 1 (Seneca, S.C.)	860	PWR	B&W	GE	Utility/Bech	Utility	100	5/71	7/73
• Oconee 2 (Seneca, S.C.)	860	PWR	B&W	GE	Utility/Bech	Utility	100	5/72	9/74
• Oconee 3 (Seneca, S.C.)	860	PWR	B&W	GE	Utility/Bech	Utility	100	6/73	12/74
McGuire 1 (Cornelius, N.C.)	1180	PWR	W	W	Utility	Utility	99	3/76	7/81
McGuire 2 (Cornelius, N.C.)	1180	PWR	W	W	Utility	Utility	86	3/77	6/83
Catawba 1 (Clover, S.C.)	1145	PWR	W	GE	Utility	Utility	76	3/79	3/84
Catawba 2 (Clover, S.C.)	1145	PWR	W	GE	Utility	Utility	18	3/80	9/85
Perkins 1 (Mocksville, N.C.)	1280	PWR	C-E	GE	Utility	Utility	0	1/81	indef.
Perkins 2 (Mocksville, N.C.)	1280	PWR	C-E	GE	Utility	Utility	0	1/82	indef.
Perkins 3 (Mocksville, N.C.)	1280	PWR	C-E	GE	Utility	Utility	0	1/83	indef.
Cherokee 1 (Gaffney, S.C.)	1280	PWR	C-E	GE	Utility	Utility	18	9/82	1/90
Cherokee 2 (Gaffney, S.C.)	1280	PWR	C-E	GE	Utility	Utility	0	9/83	1/93
Cherokee 3 (Gaffney, S.C.)	1280	PWR	C-E	GE	Utility	Utility	0	9/84	indef.
Florida Power & Light Co.									
• Turkey Point 3 (Florida City, Fla.)	666	PWR	W	W	Bechtel	Bechtel	100	8/70	12/72
• Turkey Point 4 (Florida City, Fla.)	666	PWR	W	W	Bechtel	Bechtel	100	8/71	9/73
• St. Lucie 1 (Hutchinson Island, Fla.)	777	PWR	C-E	W	Ebasco	Ebasco	100	1/73	12/76
St. Lucie 2 (Hutchinson Island, Fla.)	777	PWR	C-E	W	Ebasco	Ebasco	62	9/79	5/83

CONTINUED

NORTHEAST: Connecticut, Delaware, Maine, Maryland, Massachusetts, New Hampshire, New Jersey, New York, Pennsylvania, Rhode Island, Vermont.
SOUTH: Alabama, Arkansas, Florida, Georgia, Kentucky, Louisiana, Mississippi, North Carolina, South Carolina, Tennessee, Virginia, West Virginia.
MIDWEST: Illinois, Indiana, Iowa, Kansas, Michigan, Minnesota, Missouri, Nebraska, North Dakota, Ohio, South Dakota, Wisconsin.
SOUTHWEST: Arizona, New Mexico, Oklahoma, Texas.
WEST AND NORTHWEST: California, Colorado, Idaho, Montana, Nevada, Oregon, Utah, Washington, Wyoming.

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World List of Nuclear Power Plants, cont'd

	Net MWe	Type	Reactor Supplier	Generator Supplier	Architect Engineer	Constructor	Construction stage (%)	Commercial Operation orig. sched.†	Commercial Operation actual or expected
U.S.—SOUTH, cont'd									
Florida Power Corporation									
• Crystal River 3 (Red Level, Fla.)	825	PWR	B&W	W	Gilbert	Jones	100	9/72	3/77
Georgia Power Co.									
• Edwin I. Hatch 1 (Baxley, Ga.)	786	BWR	GE	GE	SS/Bechtel	Utility	100	4/73	12/75
• Edwin I. Hatch 2 (Baxley, Ga.)	786	BWR	GE	GE	Bechtel	Utility	100	4/76	8/79
Vogtle 1 (Waynesboro, Ga.)	1100	PWR	W	GE	SS/Bechtel	Utility	17.6	2/78	/85
Vogtle 2 (Waynesboro, Ga.)	1100	PWR	W	GE	SS/Bechtel	Utility	4.4	2/79	/88
Gulf States Utilities Co.									
River Bend 1 (St. Francisville, La.)	940	BWR	GE	GE	S&W	S&W	23	10/79	4/84
River Bend 2 (St. Francisville, La.)	940	BWR	GE	GE	S&W	S&W	0	10/81	indef.
Louisiana Power & Light Co.									
Waterford 3 (Taft, La.)	1165	PWR	C-E	W	Ebasco	Ebasco	81	1/77	3/83
Mississippi Power & Light Co.									
Grand Gulf 1 (Port Gibson, Miss.)	1250	BWR	GE	Allis	Bechtel	Bechtel	87.2	9/79	4/82
Grand Gulf 2 (Port Gibson, Miss.)	1250	BWR	GE	Allis	Bechtel	Bechtel	22.4	9/81	9/86
South Carolina Electric & Gas Co.									
Virgil C. Summer 1 (Parr, S.C.)	900	PWR	W	GE	Gilbert	Daniel	96	10/77	12/81
Tennessee Valley Authority									
• Browns Ferry 1 (Decatur, Ala.)	1067	BWR	GE	GE	Utility	Utility	100	10/70	8/74
• Browns Ferry 2 (Decatur, Ala.)	1067	BWR	GE	GE	Utility	Utility	100	10/71	3/75
• Browns Ferry 3 (Decatur, Ala.)	1067	BWR	GE	GE	Utility	Utility	100	10/72	3/77
Sequoyah 1 (Daisy, Tenn.)	1148	PWR	W	W	Utility	Utility	100	10/73	1/81
Sequoyah 2 (Daisy, Tenn.)	1148	PWR	W	W	Utility	Utility	96	4/74	7/82
Watts Bar 1 (Spring City, Tenn.)	1177	PWR	W	W	Utility	Utility	83	10/76	11/82
Watts Bar 2 (Spring City, Tenn.)	1177	PWR	W	W	Utility	Utility	70	4/77	8/83
Bellefonte 1 (Scottsboro, Ala.)	1213	PWR	B&W	BB	Utility	Utility	75	7/77	12/85
Bellefonte 2 (Scottsboro, Ala.)	1213	PWR	B&W	BB	Utility	Utility	57	4/78	9/86
Hartsville A1 (Hartsville, Tenn.)	1233	BWR	GE	BB	Utility	Utility	31	4/79	7/88
Hartsville A2 (Hartsville, Tenn.)	1233	BWR	GE	BB	Utility	Utility	23	4/80	4/89
Hartsville B1 (Hartsville, Tenn.)	1233	BWR	GE	BB	Utility	Utility	17	10/79	indef.
Hartsville B2 (Hartsville, Tenn.)	1233	BWR	GE	BB	Utility	Utility	7	10/80	indef.
Phipps Bend 1 (Surgoinsville, Tenn.)	1233	BWR	GE	BB	Utility	Utility	17	4/82	2/89
Phipps Bend 2 (Surgoinsville, Tenn.)	1233	BWR	GE	BB	Utility	Utility	5	4/83	indef.
Yellow Creek 1 (Iuka, Miss.)	1285	PWR	C-E	GE	Utility	Utility	18	4/83	4/88
Yellow Creek 2 (Iuka, Miss.)	1285	PWR	C-E	GE	Utility	Utility	3	4/84	indef.
Tennessee Valley Authority, Commonwealth Edison Co., and DOE									
Clinch River Breeder Reactor Plant (Oak Ridge, Tenn.)	350	LMFBR	W	GE	B&R	S&W	0	/80	indef.
Virginia Electric & Power Co.									
• Surry 1 (Gravel Neck, Va.)	775	PWR	W	W	S&W	S&W	100	3/71	12/72
• Surry 2 (Gravel Neck, Va.)	775	PWR	W	W	S&W	S&W	100	3/72	5/73
• North Anna 1 (Mineral, Va.)	850	PWR	W	W	S&W	S&W	100	3/74	6/78
• North Anna 2 (Mineral, Va.)	850	PWR	W	W	S&W	S&W	100	7/75	12/80
North Anna 3 (Mineral, Va.)	907	PWR	B&W	W	S&W	S&W	8.8	3/77	/89
SOUTHWEST									
Arizona Public Service Co.									
Palo Verde 1 (Wintersburg, Ariz.)	1270	PWR	C-E	GE	Bechtel	Bechtel	80.8	5/81	5/83
Palo Verde 2 (Wintersburg, Ariz.)	1270	PWR	C-E	GE	Bechtel	Bechtel	50.1	11/82	5/84
Palo Verde 3 (Wintersburg, Ariz.)	1270	PWR	C-E	GE	Bechtel	Bechtel	15.5	5/84	5/86
Houston Lighting & Power Company									
Allens Creek 1 (Wallis, Tex.)	1200	BWR	GE	GE	Ebasco	Ebasco	0	6/80	/88
South Texas Project 1 (Palacios, Tex.)	1250	PWR	W	W	Brown	Brown	58.9	10/80	/84
South Texas Project 2 (Palacios, Tex.)	1250	PWR	W	W	Brown	Brown	21.9	3/82	/86
Public Service Co. of Oklahoma									
Black Fox 1 (Inola, Okla.)	1150	BWR	GE	GE	B&V	Utility	0-lwa	4/82	7/85
Black Fox 2 (Inola, Okla.)	1150	BWR	GE	GE	B&V	Utility	0-lwa	4/84	7/88
Texas Utilities Generating Company									
Comanche Peak 1 (Glen Rose, Tex.)	1150	PWR	W	Allis	G&H	B&R	86	1/80	/82
Comanche Peak 2 (Glen Rose, Tex.)	1150	PWR	W	Allis	G&H	B&R	50	1/82	/84

• Units in commercial operation

† Estimated date of startup, announced at time reactor was ordered

cp — construction permit issued

lwa — limited work authorization issued

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	Net MWe	Type	Reactor Supplier	Generator Supplier	Architect Engineer	Constructor	Construction stage (%)	Commercial Operation orig. scheduled†	Commercial Operation actual or expected
WEST AND NORTHWEST									
Pacific Gas & Electric Co.									
• Humboldt Bay 3 (Eureka, Calif.)	63	BWR	GE	GE	Bechtel	Bechtel	100	/63	8/63
Diablo Canyon 1 (Avila Beach, Calif.)	1084	PWR	W	W	Utility	Utility	97.5	5/72	8/81
Diablo Canyon 2 (Avila Beach, Calif.)	1106	PWR	W	W	Utility	Utility	89	7/74	4/82
Portland General Electric Co.									
• Trojan (Prescott, Ore.)	1130	PWR	W	GE	Bechtel	Indep	100	9/74	5/76
Pebble Springs 1 (Arlington, Ore.)	1260	PWR	B&W	GE	Bechtel	Indep	0	7/80	indef.
Pebble Springs 2 (Arlington, Ore.)	1260	PWR	B&W	GE	Bechtel	Indep	0	6/83	indef.
Public Service Company of Colorado									
• Fort St. Vrain (Platteville, Colo.)	330	HTGR	GA	GE	S&L	GA	100	4/72	1/79
Puget Sound Power & Light Co.									
Skagit 1 (Hanford, Wash.)	1288	BWR	GE	W	Bechtel	Bechtel	0	7/81	indef.
Skagit 2 (Hanford, Wash.)	1288	BWR	GE	W	Bechtel	Bechtel	0		indef.
Sacramento Municipal Utility District									
• Rancho Seco (Clay Station, Calif.)	913	PWR	B&W	W	Bechtel	Bechtel	100	5/73	4/75
Southern California Edison and San Diego Gas & Electric Co.									
• San Onofre 1 (San Clemente, Calif.)	436	PWR	W	W	Bechtel	Bechtel	100		1/68
San Onofre 2 (San Clemente, Calif.)	1100	PWR	C-E	GEC	Bechtel	Bechtel	97	6/75	12/81
San Onofre 3 (San Clemente, Calif.)	1100	PWR	C-E	GEC	Bechtel	Bechtel	68	6/75	2/83
Washington Public Power Supply System									
• Hanford-N (Richland, Wash.)	860	LGR	GE	GE	B&R	B&R	100		7/66
WNP-2 (Richland, Wash.)	1150	BWR	GE	W	B&R	B&R	85	9/77	9/83
WNP-1 (Richland, Wash.)	1250	PWR	B&W	W	UE&C	UE&C	41	9/80	2/86
WNP-3 (Satsop, Wash.)	1240	PWR	C-E	W	Ebasco	Ebasco	23	3/82	9/86
WNP-4 (Richland, Wash.)	1250	PWR	B&W	W	UE&C	UE&C	15	3/82	2/87
WNP-5 (Satsop, Wash.)	1240	PWR	C-E	W	Ebasco	Ebasco	9	3/83	9/87

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APPENDIX B1

List of Particle Accelerators in the U. S.[10]

A directory of particle accelerators used for nuclear physics research in the United States is shown in Tables A.1 through A.8. Table A.9 lists cyclotrons whose main purpose is medical application or isotope production. Tables A.1 through A.9 constitute the most up-to-date compilation presently possible of accelerators in the United States that produce a nonnegligible radiation environment at decommissioning. However, they should not be considered to provide a comprehensive listing for planning purposes because of reference publication dates and constant advances in accelerator applications. Each machine at each location is listed with its maximum particle energy (proton or electron, unless otherwise noted), year of initial operation, and source of funding. The references and funding agencies for the information about each machine are given in Table A.10. The funding-agency indications are complete only for Department of Energy (DOE) installations. For all others, the funding agency is listed where known, otherwise ND, meaning "not DOE", is given. Injectors and boosters for larger machines are listed separately.

Table A.1. Proton Synchrotrons

Facility and Location	*Proton Energy (GeV)*	*Initial Operation*	*Funding Agency*[a]	*Reference*[a]
Zero Gradient Synchrotron[b] Argonne National Laboratory	12.7	1963	DOE	B,E
Intense Pulsed Neutron Source Argonne National Laboratory	0.5	1976	DOE	B,E
Alternate Gradient Synchrotron Brookhaven National Laboratory	33	1960	DOE	B,E
Bevatron Lawrence Berkeley Laboratory	6.2	1954	DOE	B,D,C
Fermi main ring National Accelerator Laboratory	4000	1972	DOE	B,E
Fermi booster National Accelerator Laboratory	8	1972	DOE	B

[a] *See Table A.10 for explanation of codes.*

[b] *Decommissioning scheduled to begin in October 1979.*

Table A.2. Electron Synchrotrons

Facility and Location	*Electron Energy (MeV)*	*Initial Operation*	*Funding Agency*[a]	*Reference*[a]
Ames Laboratory Iowa State University	70	1949	DOE	C
Ames Laboratory Iowa State University	70	1954	DOE	C
University of Oklahoma	70	1968	PVT	C

[a] *See Table A.10 for explanation of codes.*

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Table A.3. Positive-Ion Linacs

Facility and Location	Proton Energy (MeV)	Initial Operation	Funding Agency[a]	Reference[a]
Super-HILAC Univ. of Calif. Berkeley	85/amu	1971	DOE	C,D
LAMPF Los Alamos National Lab.	800	1972	DOE	C,D
Linac injector National Accelerator Laboratory	200	1977	DOE	A,B
Linac injector to AGS Brookhaven National Laboratory	200	1960	DOE	A,B
FMMRI Univ. of Chicago, Franklin McLean Res. Hosp.	50		DOE	A,B
Rensselear Polytechnic Inst.	100		DOE	A,B
Linac injector to ZGS Argonne National Laboratory	50	1963	DOE	A,B

[a] *See Table A.10 for explanation of codes.*

Table A.4. Electron Linacs

Facility and Location	Electron Energy (MeV)	Initial Operation	Funding Agency[a]	Reference[a]
Bates Mass. Inst. of Tech.	400	1972	DOE	C,D
Mark III S Stanford Univ.	2 000	1972	ONR/NSF	C
Mark III Stanford Univ.	1 200	1953	NSF	B
Stanford Linear Accel. Center	22 000	1966	DOE	B
Argonne National Laboratory	22	1969	DOE	B,C
Univ. of Calif. Livermore	35	1959	DOE	C
Univ. of Calif. Livermore	100	1970	DOE	C
Univ. of Chicago	50	1957	ND	C
Gulf General Atomics	100	1958	PVT	C
Univ. of Illinois	30	1970	NSF	C,D
NASA-SREL	10	1966	NSF	C
National Bureau of Standards	150	1966	DOC	C
Naval Postgraduate School	100	1965	DON[b]	C
Naval Research Laboratory	65	1964	DON	C
ORELA Oak Ridge National Laboratory	140	1969	DOE	C,D
Rensselaer Polytechnic Inst.	100	1961	DOE	C
Yale University	75	1961	DOE	C,D
Ohio State University	6		ND	A

[a] *See Table A.10 for explanation of codes.*

[b] *Performs some NSF work and currently has a DOE proposal pending.*

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Table A.5. Cyclotrons

Facility and Location	Proton Energy (MeV)	Initial Operation	Funding Agency[a]	Reference[a]
DOE owned and/or supported				
60"AVF Brookhaven National Laboratory	36	1968	DOE	B,C
88"AVF Univ. of Calif. Berkeley	60	1962	DOE	C,D
20"AVF Univ. of Calif. Livermore	15[b]	1969	DOE	C
52"AVF Univ. of Colorado	50	1962	DOE	C,A,D
31"AVF Duke University	15[b]	1968	DOE	C
76"AVF Oak Ridge National Laboratory	66	1964	DOE	C,D
Cancer Research Hospital Univ. of Chicago	16		DOE	A
Sloan-Kettering Institute	20		DOE	A
60"FF Argonne National Laboratory	23[c]	1952	DOE	A,C,D
90"FF Univ. of California	14	1955	DOE	C
50"FF Univ. of Michigan	9[c]	1936	MI	C
105"AVF Univ. of Maryland	140	1969	DOE	C,D
88"AVF Texas A & M	60	1967	DOE	C,D
Formerly DOE owned or supported				
76"AVF Univ. of Calif. Davis	55	1965	CA	C,D
50"AVF Univ. of Calif. Los Angeles	50[b]	1962	CA	C
260"AVF Univ. of Indiana	200	1975	NSF	C,A,D
67"AVF Michigan State Univ.	55	1965	NSF	C,D
76"AVF Naval Res. Laboratory	50	1967	DON	C
69"AVF Princeton University	56	1968	NSF	C,D
28"AVF Univ. of Connecticut	8	1967	PVT	C
37"AVF Oregon State Univ.	18	1969	OR	C
60"FF NASA-Lewis Research Center	21[c]	1952	NASA	C
27"FF St. Louis University	3.5[c]	1966	PVT	C
60"FF Univ. of Washington	21.5[c]	1951	WA	C

[a] *See Table A.10 for explanation of codes.*

[b] *Negative hydrogen ion.*

[c] *Deuteron.*

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Table A.6. Synchrocyclotrons

Facility and Location	*Proton Energy (MeV)*	*Initial Operation*	*Funding Agency*[a]	*Reference*[a]
184" Univ. of Calif. Berkeley	730	1946	NIH	C
Nevis Columbia University	585	1950	NSF	B,C
197" NASA-Space Radiation Effects Laboratory	660	1967	NSF	B

[a] *See Table A.10 for explanation of codes.*

Table A.7. Storage Rings

Facility and Location	*Electron Energy (MeV)*	*Initial Operation*	*Funding Agency*[a]	*Reference*[a]
SPEAR Stanford Lin. Accel. Ctr.	2500	1972	DOE	E
Tantalus I Univ. of Wisconsin	240	1968	NSF	E

[a] *See Table A.10 for explanation of codes.*

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Table A.8. Electrostatic Accelerators

Facility and Location	*Proton Energy (MeV)*	*Initial Operation*	*Funding Agency*[a]	*Reference*[a]
DOE owned and supported				
Tandem dynamitron Argonne National Laboratory			DOE	A,F
Dynamitron Argonne National Laboratory	4	1969	DOE	A
Van de Graaff Argonne National Laboratory	2		DOE	A
Pulsed-electron Van de Graaff Argonne National Laboratory	3[b]		DOE	A
Tandem Van de Graaff (FN) Argonne National Laboratory	18	1967	DOE	A,C,D
Tandem Van de Graaff Brookhaven National Laboratory	30	1970	DOE	E,C
Van de Graaff Brookhaven National Laboratory	4	1954	DOE	A
Dynamitron Brookhaven National Laboratory	3		DOE	
Tandem Van de Graaff (FN) Oak Ridge National Laboratory	13	1962	DOE	C,D
Van de Graaff Oak Ridge National Laboratory	6	1950	DOE	E,F
Van de Graaff Oak Ridge National Labotatory	3	1948	DOE	C
3-stg Tandem Van de Graaff (FN) Los Alamos Scientific Laboratory	24.5	1964	DOE	C,F
Van de Graaff Los Alamos Scientific Laboratory	9	1950	DOE	C
Van de Graaff Los Alamos Scientific Laboratory	3	1969	DOE	
Tandem Van de Graaff Yale University	22	1966	DOE	A,D
Van de Graaff Notre Dame Univ.	2		DOE	A
Tandem Van de Graaff Univ. of Calif. Livermore	27	1971	DOE	C,F

See footnotes at end of table.

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Table A.8. Continued

Facility and Location	*Proton Energy (MeV)*	*Initial Operation*	*Funding Agency*[a]	*Reference*[a]
DOE owned and supported (continued)				
3-stg Tandem Van de Graaff Duke University	32	1968	DOE	C,D,F
Van de Graaff Duke University	3.3	1961	DOE	C
Van de Graaff Duke University	4.2	1952	DOE	C
Tandem Van de Graaff Ohio University	11	1970	DOE	A,D
Van de Graaff Univ. of Chicago, Franklin McLean Res. Hosp.	2		DOE	A
Tandem Van de Graaff Kansas State Univ.	12	1969	DOE	A,C,F
Tandem Van de Graaff Univ. of Wisconsin	12	1960	DOE	A,D,F
Tandem Van de Graaff Univ. of Minnesota	20	1966	DOE	A,D,F
Not DOE owned, but formerly DOE supported				
Van de Graaff Johns Hopkins Univ.	3	1963	DOE	A
Van de Graaff Columbia University	5.5	1955	COL	C
Tandem Van de Graaff Rice University	12	1961	ND	C,D,F
Van de Graaff Rice University	5.5	1953	ND	C,F
Tandem Van de Graaff West Michigan Univ.	12	1969	SMI	C,F
Tandem Van de Graaff Purdue University	15	1969	SIN	C,F
Van de Graaff Univ. of Kansas	4	1963	ND	C

See footnotes at end of table.

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Table A.8. Continued

Facility and Location	*Proton Energy (MeV)*	*Initial Operation*	*Funding Agency*[a]	*Reference*[a]
Not DOE owned, but formerly DOE supported (continued)				
Van de Graaff Univ. of Maryland	3.5	1958	ND	C
Tandem Van de Graaff Florida State Univ.	18	1970	ND	C,F
Van de Graaff Brigham Young Univ.	2.5	1965	BYU	C
Van de Graaff Brigham Young Univ.	4	1970	BYU	C
Van de Graaff Case Western Reserve Univ.	3	1958	ND	C
3-stg Tandem Van de Graaff, (CN & EN) Univ. of Texas - Austin	17.5	1963	STX	C,D,F
Van de Graaff Univ. of Texas - Austin	4	1963	STX	C
Not DOE owned and never DOE supported				
3-stg Tandem Van de Graaff (EN & EN) Univ. of Pittsburgh	18	1967	NSF	C,D,F
3-stg Tandem Van de Graaff (FN & FN) Univ. of Washington	25.5	1967	SWA	C,D,F
Tandem Van de Graaff (T-8) Aerospace Res. Lab.	8	1967	USAF	C
Van de Graaff Aerospace Res. Lab.	2	1953	USAF	C,F
Tandem Van de Graaff (FN) Army Nuclear Eff. Lab., Edgewood Arsenal	15	1969	USA	C,F
Tandem Van de Graaff (EN) Calif. Inst. of Tech.	10	1961	NSF	C
Tandem Van de Graaff (FN) State Univ. of New York, Stony Brook	17	1968	NSF	C,D,F

See footnotes at end of table.

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Table A.8. Continued

Facility and Location	*Proton Energy (MeV)*	*Initial Operation*	*Funding Agency*[a]	*Reference*[a]
Not DOE owned and never DOE supported (continued)				
Tandem Van de Graaff Notre Dame Univ.	15	1968	NSF	C,D,F
Van de Graaff Notre Dame Univ.	4	1956	ND	C
Tandem Van de Graaff (EN) Univ. of Pennsylvania	12	1962	NSF	C,F
Tandem Van de Graaff (MP) Univ. of Rochester	20	1966	NSF	C,D,F
Tandem Van de Graaff (FN) Rutgers University	18	1964	NSF	C,D,F
Tandem Van de Graaff (FN) Stanford University	19	1965	NSF	C,D,F
Van de Graaff Stanford University	3	1960	NSF	C
Van de Graaff (CN) Univ. of Arizona	5.5	1968	NSF	C,F
Van de Graaff (CN) Bartol Res. Found.	5.5	1952	PVT	C,F
Van de Graaff Univ. of Georgia	5	1970	ND	C
Van de Graaff (CN) Univ. of Iowa	6	1964	ND	C,F
Van de Graaff Univ. of Iowa	2	1961	ND	C
Van de Graaff (CN) Univ. of Kentucky	6	1964	NSF	C,F
Van de Graaff (CN) Lowell Tech. Inst.	6	1969	PVT	C,F
Original Van de Graaff Mass. Inst. of Tech.	9	1951	PVT	C
Van de Graaff (CN) Naval Research Lab.	5.5	1953	DON	C
Van de Graaff Oregon University			NSF	D
Van de Graaff (CN) Ohio State Univ.	6	1963	ND	C,F

See footnotes at end of table.

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Table A.8. Continued

Facility and Location	*Proton Energy (MeV)*	*Initial Operation*	*Funding Agency*[a]	*Reference*[a]
Not DOE owned and never DOE supported (continued)				
Van de Graaff (CN) Penn. State Univ.	5.5	1963	ND	C,F
Van de Graaff (CN) Univ. of Virginia	6	1966	ND	C,F
Homemade Van de Graaff Calif. Inst. of Tech.	0.6	1952	ND	C
Homemade Van de Graaff Calif. Inst. of Tech.	2.8	1949	ND	C
Van de Graaff Carnegie Inst.	4	1938	ND	C
Van de Graaff Univ. of Florida	4.2	1964	ND	C
Van de Graaff Georgetown Univ.	0.4	1964	NSF	C
Van de Graaff Georgetown Univ.	2	1966	NSF	C
Van de Graaff Lockheed Corp.	3	1958	ND	C
Van de Graaff Montana State Univ.	3.5	1971	ND	C
Dynamitron NASA-SREL	3[b]	1966	NASA	C
Van de Graaff Nat. Bur. of Standards	2	1955	DOC	C
Dynamitron State Univ. of New York, Albany	4	1970	SUNY	C
Van de Graaff Univ. of Oregon	4.5	1966	NSF	C
Van de Graaff Univ. of South Carolina	0.5	1962	ND	C
Van de Graaff Chicago Nuclear (formerly Texas Nuclear)	3.2	1956	PVT	C
Van de Graaff Tulane Univ.	3	1967	PVT	C

See footnotes at end of table.

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Table A.8. Continued

Facility and Location	Proton Energy (MeV)	Initial Operation	Funding Agency[a]	Reference[a]
Not DOE owned and never DOE supported (continued)				
Van de Graaff Virginia Polytechnic Inst.	4	1968	ND	C
Van de Graaff Washington State Univ.	2	1965	ND	C
Van de Graaff West Virginia Univ.	2[c]	1964	ND	C
Van de Graaff Worchester Polytechnic Inst.	2	1959	ND	C

[a] *See Table A.10 for explanation of codes.*

[b] *Electron.*

[c] *Deuteron.*

Table A.9. Medical-Application Cyclotrons

Facility and Location	Proton Energy (MeV)	Initial Operation	Funding Agency[a]	Reference[a]
CS15(20") Washington Univ. Medical School	15	1976		G
54"AVF Mallinckrodt Inst. of Radiology	13	1965		C,D
Allis Chalmers Massachusetts Genl. Hospital				G
CS22(29") New England Nuclear	22		PVT	G
CS30(40") New England Nuclear	30	1976	PVT	G
CS22(29") Medi-Physics	22		PVT	G
CS22(29") Medi-Physics	22		PVT	G
CS15(20") Univ. of Chicago-Franklin McLean Res. Hosp.	15			G
CS30(40") Mt. Sinai Hospital	30		PVT	G
CS22(29") UCLA Medical School	22		PVT	G

[a] *See Table A.10 for explanation of codes.*

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Table A.10. Codes for Funding Agencies and References

Code	*Funding Agency - Reference*
BYU	Brigham Young University
CA	University of California
COL	Columbia University
DOC	Department of Commerce
DOE	Department of Energy
DON	Department of the Navy
MI	University of Michigan
NASA	National Aeronautics and Space Administration
ND	Not-DOE funded
NIH	National Institutes of Health
NSF	National Science Foundation
ONR	Office of Naval Research
OR	Oregon State University
PVT	Private funding
SIN	State of Indiana
SMI	State of Michigan
STX	State of Texas
SUNY	State University of New York
SWA	State of Washington
USA	United States Army
USAF	United States Air Force
WA	University of Washington

A "AEC Owned Accelerators." Provided by J.J. Nelsen, USDOE/CORO/OES, 1978.

B "American Accelerators Above 200 MeV." Provided by E.T. Ritter, USDOE, Div. Physical Research, 1978.

C "Physics in Perspective." Vol. II, Part A (The Bromley Report) NAS, 1969. Table 11-5: Census of Accelerators in Basic Nuclear Physics Research.

D "Future of Nuclear Science." (The Friedlander Report) NAS, 1976. App. G: Accelerators Funded for Nuclear Science.

E "Catalogue of High Energy Accelerators." 16th International Conference on High Energy Accelerators, SLAC, May 1974.

F "The Development of Electrostatic Accelerators." Nuclear Instruments and Methods, Vol. 122, Nov-Dec 1974. Tables 1-4.

G A.P. Wolf. "Medical Cyclotrons." IAEA Symposium on Medical Radionuclide Imaging, BNL 22021, CONF-761060-10, Oct 1975.

APPENDIX B2

List of Decommissioned Accelerators[10]

Institution	*Identification*	*Disposition*	*Date*
Brookhaven Nat'l Lab.	Cosmotron (3-GeV Proton Synchrotron)	Dismantled	1966
Brookhaven Nat'l Lab.	18" FF Cyclotron	Dismantled	1963
Univ. Cal., Los Angeles	41" Synchrocyclotron	Dismantled	1961
Univ. Cal., Los Angeles	50" AVF Cyclotron	Dismantled	1973
Cal. Inst. of Tech.	1.1-GeV Electron Synchrotron	Dismantled	1968
Carnegie Inst.	60" FF Cyclotron	Stored	1955
Carnegie Mellon Univ.	141" Synchrocyclotron	Dismantled	1973
Univ. of Chicago	170" Synchrocyclotron	Partially dismantled	
Univ. of Chicago	100-MeV Betatron	Dismantled	1955
Univ. of Chicago	32" FF Cyclotron	Dismantled	1944
Univ. of Chicago	170" FM Cyclotron	Dismantled	1971
Chicago Nuclear	3-MV Van de Graaff	Shut down	1972
Columbia Univ.	37" FF Cyclotron	Dismantled	1965
Columbia Univ.	5.5-MV Van de Graaff	Shut down	1970
Cornell Univ.	Cornell Electron Synchrotron (CES)	Dismantled	
Univ. of Florida	1-MV Van de Graaff	Dismantled	1965
General Electric Co., Schenectady	100-MeV Betatron	Dismantled	1973
Harvard Univ.	Cambridge Electron Accelerator (CEA)	Dismantled	1973
Univ. of Illinois	50" FF Cyclotron	Dismantled	1968
Univ. of Illinois	300-MeV Betatron	Dismantled	1968
Univ. of Illinois	25-MeV Betatron	Dismantled	1972
Indiana Univ.	45" FF Cyclotron	Dismantled	1968
Iowa State Univ.	70-MeV Electron Synchrotron	Mothballed	1972
Iowa State Univ.	4-MV Van de Graaff	Dismantled	1964
Johns Hopkins Univ.	1-MV Van de Graaff	Dismantled	1962
Johns Hopkins Univ.	3-MV Van de Graaff	Dismantled	1974
Knolls Atomic Power Lab., Schenectady	100-MeV Betatron	Scrapped	1970
Univ. of Texas	2-MV Van de Graaff	Dismantled	1962
Lawrence Lab., Berkeley	340-MeV Electron Synchrotron	To Smithsonian	1960
Lawrence Lab., Berkeley	32" FF Cyclotron	To Lawrence Museum	1940
Lawrence Lab., Berkeley	27" FF Cyclotron, 6 MeV	To museum	
Lawrence Lab., Berkeley	30-MeV Proton Linac	Decommissioned	1960
Lawrence Lab., Livermore	90" FF Cyclotron	Dismantled	1971
Lawrence Lab., Livermore	35-MeV Electron Linac	Dismantled	1970
Los Alamos Sci. Lab.	2.5-MV Van de Graaff	Dismantled	1965
Los Alamos Sci. Lab.	2.5-MV Van de Graaff	Dismantled	1963
Los Alamos Sci. Lab.	FF Cyclotron	Dismantled	1974
Los Alamos Sci. Lab.	Electron Prototype Accelerator (EPA)	Shut down (?)	~1970
Los Alamos Sci. Lab.	15-MeV Betatron	Junked	~1970
Los Alamos Sci. Lab.	10-MeV "Portable" Betatron (10 units)	Junked	1955
Los Alamos Sci. Lab.	10-MeV "Portable" Betatron	Junked	1955
Los Alamos Sci. Lab.	10-MeV "Portable" Betatron, modified	To Univ. of N.M.	1955
Louisana State Univ.	1-MV Van de Graaff	Dismantled	1966
Mass. Inst. of Tech.	42" FF Cyclotron	Mothballed	1973
Mass. Inst. of Tech.	300-MeV Electron Synchrotron	Dismantled	1960
Mass. Inst. of Tech.	2-MV Van de Graaff	To Boston Mus. of Sci.	1950
Mass. Inst. of Tech.	17-MeV Electron Linac	Dismantled	1968
Mass. Inst. of Tech.	8-MV Van de Graaff	Stored	1970
Mass. Inst. of Tech.	4-MV Van de Graaff	Dismantled	1960
Univ. of Michigan	100-MeV Electron Synchrotron	Dismantled	1960
Univ. of Michigan	83", 35-MeV AVF Cyclotron	Dismantled	1978
Univ. of Michigan	50" FF Cyclotron	Mothballed	1970
Univ. of Minnesota	68-MeV Proton Linac	Dismantled	1968
Univ. of Minnesota	4-MV Van de Graaff	Dismantled	1966
Nat'l Bureau of Stds.	50-MeV Betatron	--	1965

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Institution	*Identification*	*Disposition*	*Date*
Naval Research Lab.	2-MV Van de Graaff	Dismantled	1966
Northwestern Univ.	4.5-MV Van de Graaff	Stored	1968
Oak Ridge Nat'l Lab.	63" FF Cyclotron	To museum	1962
Ohio State Univ.	2-MV Van de Graaff	Dismantled	1962
Ohio State Univ.	45" FF Cyclotron	Shut down	1972
Oregon State Univ.	37" AVF Cyclotron	Dismantled	1973
Univ. of Pennsylvania	12-MV Tandem Van de Graaff (EN)	Dismantled	1974
Penn. State Univ.	6-MV Van de Graaff, CN	Shut down	1973
Univ. of Pittsburgh	45" FF Cyclotron	Dismantled	1967
Princeton Univ.	3-GeV Proton Synchrotron (PPA)	Sent to ANL	
Princeton Univ.	35" Synchrocyclotron	Mothballed	1967
Purdue Univ.	37" FF Cyclotron	Dismantled	1968
Purdue Univ.	300-MeV Electron Synchrotron	Dismantled	1961
Rensselaer Poly. Inst.	30-MeV Betatron	Dismantled	1968
Univ. of Rochester	130" Synchrocyclotron	Dismantled	1963
Univ. of Rochester	240-MeV Electron Synchrotron	Stored	1968
Univ. of S. California	31-MeV Proton Linac	Dismantled	1967
Space Radiat. Eff. Lab.	Synchrocyclotron	Mothballed	1979
Stanford Univ.	Mark I Electron Linac	Extended to Mark II	1948
Stanford Univ.	Mark II 1200-MeV Electron Linac	Sent to Sao Paulo, Brazil	1949
Stanford Univ.	Mark III Electron Linac	Dismantled, parts replaced	1963
Univ. of Texas, Austin	17.5-MV 3-St. Tandem VdG (CN + EN)	Mothballed	1977
Tulane Univ.	3-MV Van de Graaff	Sold	1973
Univ. of Virginia	70-MeV Electron Synchrotron	Stored	1968
Washington State Univ.	2-MV Van de Graaff	Shut down	1973
Yale Univ.	HILAC	Dismantled	1974

Sources: "Nuclear Science: A Survey of Funding, Facilities and Manpower," NAS, 1975; "Physics in Perspective," Vol. II Pt. A, NAS, 1969; plus contributions from reviewers who helped update the list to mid-1978.

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APPENDIX C

List of Propulsion Reactors (Maritime)[15]

Propulsion (Maritime)

Name and/or owner	Nuclear designer	Shipbuilder	Type	Maximum shaft horsepower	Power, kW(t)	Start-up	Shut-down
SHUT DOWN							
Nuclear Ship SAVANNAH (Maritime Administration)	B&W	NYSC	Pressurized water	22,000	80,000	1961	1971

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This inventory of high level nuclear waste in the U.S. was developed from existing data, reports, and other sources of information. Existing and projected inventories of high level waste (HLW), transuranic (TRU) waste, spent fuel, and decommissioning and dismantling (D&D) waste will be required in making prudent decisions in the licensing of high level waste repositories.

Numerous references, reports, and data sources were consulted for this work. Discussions with responsible individuals at the major waste storage facilities were held to help resolve and understand discrepancies that appear in the literature. Low level wastes are excluded from this inventory. For decommissioning and dismantling wastes, no detailed projection of future wastes could be developed although rough D&D waste projections for commercial power reactors are available.

The inventory is reported in three parts. Part 1 gives the existing and projected inventory of HLW, spent fuel, and TRU wastes. Part 2 is concerned with the characteristics and volume of high activity D&D waste from commercial power reactors. Part 3 provides additional information on D&D wastes from power reactors and particle accelerators.

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